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ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Radiation Protection II, contains five modules: Film Dosimetry; Thermoluminescent Dosimetry; Radiation Dose Assessment; Personnel Dosimeters; Internal Dose Assessment; and Bioassay Techniques and Recordkeeping for Radiation Protection. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-643.) (LRA)

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ED192099

Nuclear Technology Series

COURSE 17
RADIATION PROTECTION II

U S DEPARTMENT OF HEALTH, EDUCATION & WELFARE
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AUG 1980

PREFACE

ABOUT NUCLEAR TECHNOLOGY MODULES

The modules – developed for use in two-year postsecondary technical institutions to prepare technicians for employment – are useful in industry for updating employees in company-sponsored training, and as laboratory texts in engineering schools. The principles, techniques and skills taught in the modules, based on tasks that nuclear technicians perform, were obtained from a nationwide advisory committee of employers of nuclear technicians. Each module was written by a working technical expert/experienced writer team and tested with students.

A module contains the following elements:

Introduction, which identifies the topic and often includes a rationale for studying the material.
Module Prerequisites, which identify the material a student should be familiar with before studying the module.
Objectives, which clearly identify what the student is expected to know for satisfactory module completion. The objectives, stated in terms of action-oriented behaviors, include such action words as operate, measure, calculate, identify and define, rather than words with many interpretations, such as know, understand, learn and appreciate.
Subject Matter, which presents the background theory and techniques supportive to the objectives of the module. Subject matter is written with the technical student in mind.
Exercises, which provide practical problems to which the student can apply this new knowledge.
Materials, which identify the equipment required to complete the laboratory procedure.
Procedures, which is the experiment section, or "hands-on" portion of the module (including step-by-step instruction) designed to reinforce student learning.
Data Tables, which are included in most modules for the first year (or basic) courses to help the student learn how to collect and organize data.
Reference Materials, which are included as suggestions for supplementary reading/viewing for the student.
Test, which measures the student's achievement of the prestated objectives.

A helpful Index has been added at the end of each course.

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The Nuclear Technology Series consists of five curricula:

Nuclear Reactor (Plant) Operator Trainee.
Radiation Protection Technician.
Nuclear Materials Processing Technician.
Nuclear Quality-Assurance/Quality-Control Technician.
Nuclear Instrumentation and Control Technician.

The completed Nuclear Technology Series consists of the following 35 technical specialty courses:

Radiation Physics.
Radiation Protection I.
Principles of Process Instrumentation.
Heat Transfer and Fluid Flow.
Introduction to Nuclear Systems.
Instrumentation and Control of Reactors and Plant Systems.
Reactor Operations.
Reactor Safety.
Reactor Auxiliary Systems.
Power Plant Systems.
Radiation Detection and Measurement.
Reactor Physics.
Power Plant Chemistry.
Introduction to Quality Assurance/ Quality Control.
Metallurgy and Metals Properties.
Mechanical Component Characteristics and Specifications.
Radiation Protection II.
Radiological Emergencies.
Radiation Shielding.
Radiation Monitoring Techniques (Radiochemical).
Radioactive Materials Disposal and Management.
Advanced Radionuclide Analysis.
Nuclear Chemical Processes.
Nuclear Systems and Safety.
Radioactive Material Handling Techniques.
Nondestructive Examination (NDE) Techniques I.
Metrology.
Welding Inspection.
Civil/Structural Inspection.
Mechanical Inspection.
Quality-Assurance Practices.
Nondestructive Examination (NDE) Techniques II.
Control Systems I.
Control Systems II.
Power Plant Electrical Systems and Components.

Availability: Instructional modules are available individually or by courses from:

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Printed:

COURSE MODULES

COURSE 17 RADIATION PROTECTION II

MODULE 17-1	Film Dosimetry
MODULE 17-2	Thermoluminescent Dosimetry (TLD)
MODULE 17-3	Radiation Dose Assessment - Personnel Dosimeters
MODULE 17-4	Internal Dose Assessment
MODULE 17-5	Bioassay Techniques and Recordkeeping for Radiation Protection

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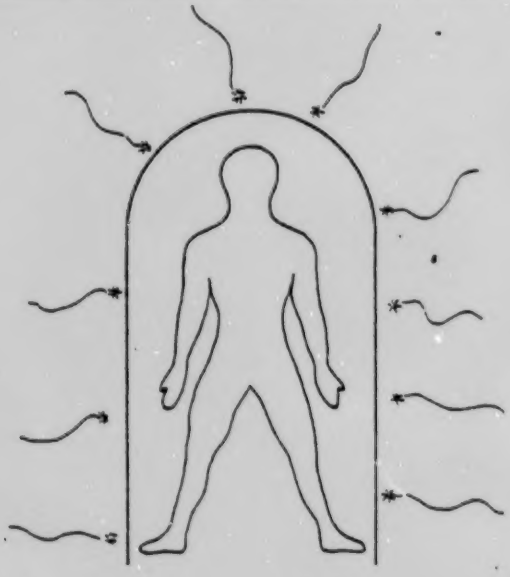
NUCLEAR TECHNOLOGY

MODULE 17 - 1

FILM DOSIMETRY



RADIATION PROTECTION II

COURSE 17

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 17 RADIATION PROTECTION II

MODULE 17-1 Film Dosimetry

MODULE 17-2 Thermoluminescent Dosimetry (TLD)

MODULE 17-3 Radiation Dose Assessment - Personnel Dosimeters

MODULE 17-4 Internal Dose Assessment

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INTRODUCTION

Film dosimetry is the oldest, and still a widely used, system of personnel dosimetry. It first was used in 1903 by an American dermatologist, who placed film on a patient that he was X-raying to determine whether or not the patient was receiving too large an exposure. The first use of film badges was by radiologist George Pfahler, who in 1922 advocated carrying film in the breast pocket as a routine procedure for radiation workers. The film would be developed bi-weekly to determine how much fogging had occurred. He also suggested that the use of a paper clip or penny over the film would tell, to some extent, how much radiation exposure the individual had incurred. The differential blackening of the film, he felt, would give at least a rough approximation.

In 1926, Edith Quimby, a radiological physicist, devised the first film badge having filters. The filters were utilized to correct for the energy response of the film, thus permitting more accurate measurements to be made. Robert S. Landauer, Sr., another physicist, made a badge similar to that of Quimby's, in the same year, and placed dental X-ray packets inside. These were the first crude film badges.

In 1928, Hans Franke was able to correlate the degree of film blackening with the exposure in units of roentgens. This correlation allowed scientists to measure personnel doses with film.

Soon, film dosimetry was used extensively. Scientists in the Manhattan project greatly improved the film badge and employed it not only for beta and gamma radiations, but to measure the dose from thermal and fast neutrons. Film badges were used in the project because they were inexpensive and light-weight; the technology was developed; and film was easy

to acquire. They also provided a method of recording a wide range of radiation doses in the region of interest and a permanent record, to which reference could be made as some later time.

MODULE PREREQUISITES

The student should have completed courses in basic chemistry and algebra, Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; and Course 11, "Radiation Detection and Measurement."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define the following terms:
   - Latent image.
   - Optical density.
   - Solarization.
   - Sensitivity.
   - Energy dependence.
   - Radiophotoluminescent.
   - Film calibration.
2. Describe the theoretical basis of film dosimetry.
3. Evaluate the effects of energy dependence on the interpretation of a beta-gamma film badge.

4. Evaluate the effects of environmental conditions such as temperature and humidity upon the response and interpretation of a film badge.
5. Describe how nuclear emulsions can be used to measure neutron dose.
6. Given the appropriate equipment, make a calibration curve for a film badge, basing the curve on actual exposure, development, and readout for the dose range 20-mrad to 1 rad for a given source of radiation. Plot the results on log-log paper, and label the curve at the appropriate points.
7. Given the appropriate equipment, expose 10 film packs to the same dose of radiation; develop simultaneously; and make a table of the optical densities.
8. Expose a series of film packs to the same dose of radiation, and vary time and temperature during development. Make a plot of the density obtained versus temperature and time.
9. Given the appropriate equipment, make an autoradiograph of a radioactive labeled substance.

FILM FOR RADIATION DOSIMETRY

A photographic film used for radiation dosimetry basically consists of an emulsion laid on a film base, usually a nonflammable substance such as cellulose acetate. The film emulsion contains grains of silver halides, usually silver bromide, ranging from about a tenth of a micrometer to ten micrometers in diameter. The grains are sensitive to radiation exposure. The larger the grain size, the more sensitive the film is to radiation.

The base is relatively thin, perhaps only a few micrometers in thickness. The emulsion, in a net form, is placed over this base and allowed to dry. Depending upon the use of the film, the emulsion may range in thickness from a few micrometers to several hundreds of micrometers. Most films for beta-gamma monitoring are only a few tenths of a micrometer thick, although film designed for the monitoring of fast neutrons is much thicker, normally on the order of a few hundred micrometers. The emulsion is overlaid with a very thin protective coating called the "T" coat, and then wrapped in black paper to prevent light from entering.

When the film is exposed to radiation, the grains of silver bromide absorb energy. A few of the silver atoms in the crystal structure of the grain are reduced chemically to metallic silver, according to the relationship –

$$Ag^{+} + e^{-} \longrightarrow Ag^{o}.$$

Of the many tens of thousands of silver atoms in the crystal, only a very few are directly reduced to metallic silver by exposure to radiation. The remainder are unchanged; if a person were to observe an exposed film, it would appear to be identical to an unexposed film. Thus, these few atoms of atomic or metallic silver are known as the latent (hidden) image, which is brought out, or amplified, when the film is developed.

Development of the film is simply a matter of chemical treatment to amplify the latent image. The film is dipped into a chemical reducing agent, usually an organic chemical such as metal or hydroquinone, which quickly reduces the silver in the silver halide grains to metallic silver.

The already reduced silver atoms of the latent image serve as centers upon which the development chemical can act; the atoms act as catalysts to expedite this process. The film is dipped in the developing solution only long enough to allow the silver reduced by the radiation to act as catalysts. If allowed to remain in the developer too long, the unexposed silver halide grains will be reduced also; and the film would not be an accurate indicator of radiation exposure. Thus, the time in the developer is rigidly controlled. The temperature of the development is also controlled rigidly, for the rate of the chemical reactor proceeds faster in warmer solutions. Normally, development is carried out at a temperature of 20°C (68°F) for approximately 3 to 5 minutes.

To ensure that the development process is precise and uniform, the developer must be kept fresh because the active chemical fraction is depleted with use. Moreover, the solution is mixed or agitated when the films are being developed.

Agitation can be accomplished either mechanically, or, as frequently done, by the bubbling nitrogen of gas through the developer solution.

After the film has been developed, it is washed off in a water or chemical bath, known as the "stop bath," because it stops the action of the developer. This process normally is brief – only a minute or so. The films then are fixed by being held in a chemical bath for a time, usually about 15 or 20 minutes. In fixing procedure, chemicals such as sodium thiosulfate and sodium metabisulfite dissolve away the undeveloped silver halide grains, leaving behind only the reduced silver. When the film is washed in clean water and dried, the development procedure has ended. The film then can be read or evaluated to determine the exposure. The development process is shown schematically in Figure 17-1-1. Since film

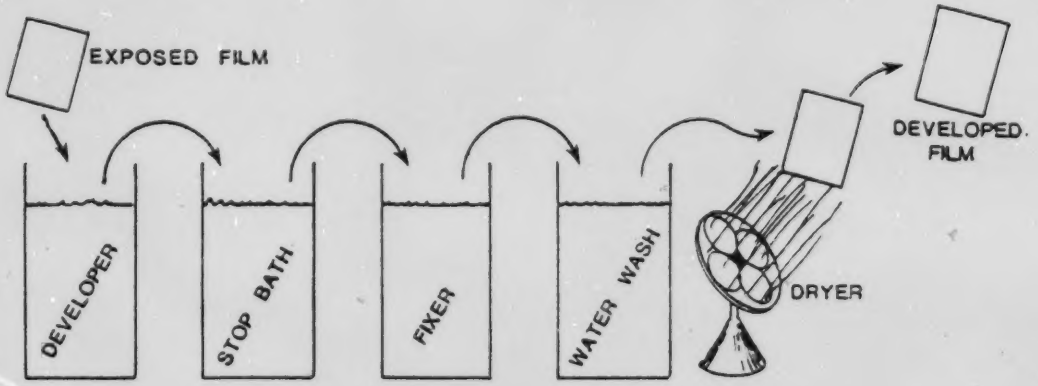

Figure 17-1-1. Schematic representation of film development.

is sensitive to light, development is carried out in the dark – either in complete darkness or in a tiny amount of red light. After development, the film can be brought safely into bright light.

Development is essentially an amplification process in which the reduced silver atoms of the latent image are multiplied many millions of times to produce a layer of free metallic silver on the film base. This layer of silver produces the characteristic blackening of the developed film and causes its optical density to increase. Optical density (OD) is a measure of how much light can pass through the film; mathematically, it is expressed by Equation 17-1-1.

$$OD = \log \left[\frac{I_o}{I}\right] \qquad \text{Equation 17-1-1 (Equation = Eq.)}$$

where:

I_o = total amount of the light that the undeveloped film permits to be transmitted through it without any exposure.

I = amount of light that the film permits to pass through it after development.

OD = optical density.

The total light that passes through the film is measured from thin beam of light allowed to pass through the film. The beam of light then is measured by a photocell connected to an ammeter (Figure 17-1-2). The measuring instrument is called a "densitometer," and reading of the film is known as "sensitometry."

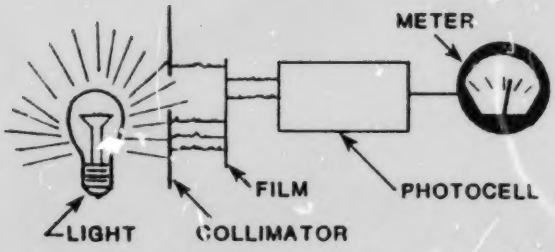

Figure 17-1-2. Schematic sketch of a densitometer.

Even an unexposed film will experience some light absorption, or "background density," which must be subtracted to obtain the "net optical density" (N.O.D.). N.O.D. is the amount of blackening attributable to the radiation dose; it is calculated by subtracting out the background density, I_{Bkgd}, as in Equation 17-1-2.

$$N.O.D. = \log\left[\frac{I_o}{I}\right] - \log\left[\frac{I_o}{I_{Bkgd}}\right] \qquad \text{Eq. 17-1-2}$$

Or, the subtraction indicated in Equation 17-1-2 can be performed, resulting in the simpler form in Equation 17-1-3.

$$N.O.D. = \log\left[\frac{I_{Bkgd}}{I}\right] \qquad \text{Eq. 17-1-3}$$

As indicated by Equation 17-1-3, only the background, or control-film density, and the dosed (unknown) film density need be obtained.

RESPONSE CHARACTERISTICS OF FILM

The optical density of an exposed and developed photographic film is a function of the radiation dose, as depicted in Figure 17-1-3. This characteristic curve for the response of a photographic emulsion consists of five smoothly joined segments. In the first, from A_0 to A, the density does not increase appreciably with the dose, and the film is not usable in this region. From A to B, the density is proportional to dose, and the film becomes useful for dosimetry. The region AB is known as the "toe" of the curve. The next region, from B to C, is the largest; and in this region, the density is proportional to the logarithm of the dose. This

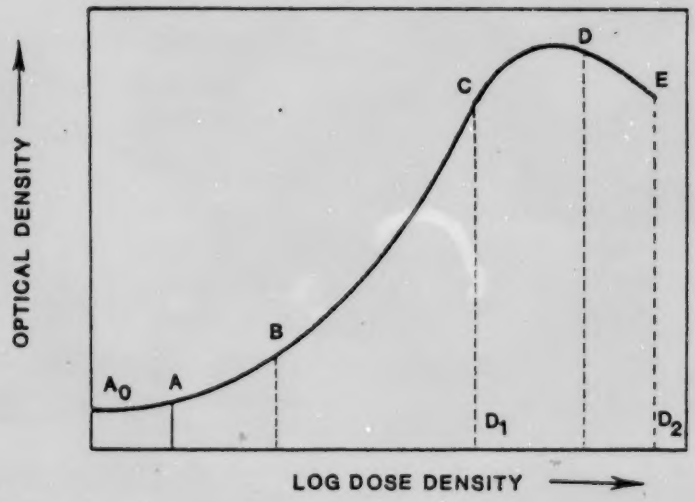

Figure 17-1-3. Characteristic response curve of a photographic film exposed to ionizing radiation.

is the region in which film is most accurate and useful for dosimetry. The region bounded by CD is the shoulder of the curve, where density hardly increases, reaching a peak and

then beginning to drop off as indicated by the tail from D to E. The reason for the apparent reversal in sensitivity is complex, and the phenomenon is named "solarization."

If solarization occurs, the density reading at S_o could be interpreted as being a dose of D_1 or D_2; therefore films must be interpreted carefully to avoid errors due to solarization. One way of minimizing such errors is to include a second piece of film with a broader range that overlaps; another way is to use a film with low solarization potential. A skilled film interpretor, however, usually can identify solarization by other response characteristics. Since doses normally are quite high, and investigation by health physics personnel also can shed light on the exposure. The type of film response curve in Figure 17-1-3 sometimes is known as an "H and D curve," named after two men, Hurter and Driffeld, who discovered it.

The density or light transmission of the film is determined by a densitometer. The densitometer is an instrument that contains a strong, focussed light source directed onto a photocell. With no film in place, light transmission is 100%; with an unexposed film, somewhat less; and with an exposed film, still less. A densitometer usually does not read the percent of light passing through, but directly reads the optical density. Conversion to % transmitted light, or vice-versa, can be easily calculated from Equation 17-1-1, as in the following example:

EXAMPLE

Given: A film has an optical density of 2.

Find: The percent of the light passes through it.

Solution:

$$O.D. = \log \left[\frac{I_o}{I}\right] \qquad \text{Eq. 17-1-1}$$

Rearrange terms:

$$O.D. = \log I_o - \log I.$$

and substitute O.D. = 2, giving

$$2 = \log I_o - \log I. \qquad \text{Eq. 17-1-4}$$

There are still two unknowns in Equation 17-1-4; however, I_o is the amount of light transmitted with no film and, hence, is equal to 100%. This can be substituted into Equation 17-1-4 to give –

$$2 = \log(100) - \log I,$$

and

$$2 = 2 - \log I.$$

Hence, $\log I = 0$,

and $I = 10^0 = 1\%$.

Most densitometers are equipped with three-decade logarithmic meters that permit readout from about 0.01 to 3 or 4, which is the upper range of most instruments. A few may read to a density of 6, but these are quite expensive and normally utilized for research rather than for routine personnel dosimetry. Developed unexposed film typically will read a

few hundredths to a few tenths of a density unit; indeed, sometimes it is possible to adjust the densitometer and the development time in such a manner that 0.01 density units = 0.01 rad (i.e. mrad) and that 0.1 density units is approximately 0.1 rad.

The sensitivity of the film is the amount of blackening or optical density produced by a given radiation exposure. Sensitivity has a precise physical definition; namely, the reciprocal of the exposure required to produce a density of 0.30. Units of sensitivity, therefore, are $\frac{1}{R}$ or R^{-1}. The sensitivity of a photographic film can be controlled in many ways – but chiefly by variation of the grain size. A larger grain size will result in a more sensitive film, since only one reduced silver atom in a grain (the latent image) is required to cause development of the entire grain. The larger the grain size, therefore, the greater the probability that a given dose will produce a latent image of one or more reduced Ag atoms in it. Sensitivity also is regulated by emulsion thickness, by the addition of certain sensitizing agents to the emulsion or silver bromide grain, or by a coating on one or both sides of the film base.

Typically, photographic films are designed to have a sensitivity of about 5 R^{-1}; that is, an exposure of about 0.2 will produce a density of 0.3. This sensitivity provides a useful range of about 100 mR to perhaps a few R (Figure 17-1-4).

A second film, covering the range from about 1 R to perhaps 800 R, may also be included. Sometimes, a film base may be coated on one side with a high-sensitivity emulsion and on the other with a low-sensitivity emulsion, thus combining both features and, hence, a very wide range into one piece of film.

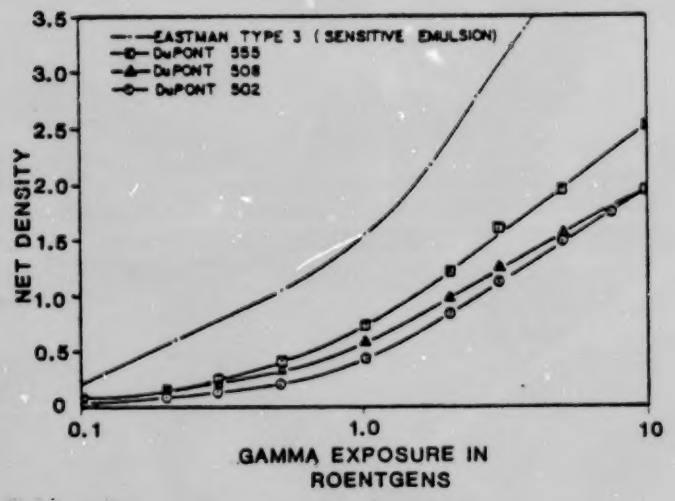

Figure 17-1-4. Density vs. dose relationships for four personnel dosimetry films.

The high-range (i.e., the low-sensitivity) emulsion is ready after the darkened low-range emulsion is stripped from the film base.

There are many drawbacks and difficulties in the use of films, one difficulty being the phenomenon of energy dependence. Energy dependence is a result of the high-Z components in film, primarily the silver (Z = 48) and bromine (Z = 35) that make up the sensitive portion of the emulsion.

The photoelectric effect predominates at low energies as the mechanism of photons' interactions with matter. The probability of the photoelectric effect is approximately proportional to Z^5; the high effectiveness of Z of the emulsion (relative to air: Z_{eff} = 8.14) produces a pronounced overresponse for photons with energies below about 400 keV (Figure 17-1-5). The density produced by 1-R of 30-keV photons could be 30 to 40 times the density produced by 1 R of photons (gamma rays) from ^{60}Co, with an effective energy of 1250 keV.

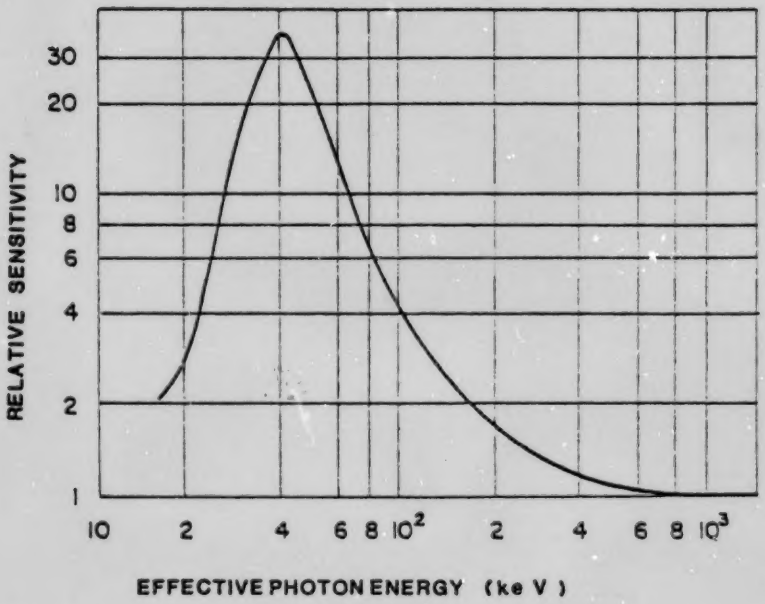

Figure 17-1-5. Typical energy dependence curve for a personnel monitoring film.

This problem may be minimized by appropriate filters. Usually, three filters are utilized, depending upon what radiations need to be eliminated. All of the beta rays and most of the low-energy X or gamma rays, can be eliminated by filters constructed of a material of fairly high atomic number, such as lead, tantalum, or cadmium. This type of filtering will make the film response reasonably flat; that is, the amount of blackening per roentgen of exposure will be equal for all energies from 80 or 100 KeV on up to 2 or 3 MeV (Figure 17-1-6). Other filters can be used to discriminate against β rays and to evaluate the lower energy photons. Ideally, four filters should be sufficient for all cases, as indicated in Table 17-1-1. The low-Z filter (lucite in the example in Figure 17-1-6) reading is subtracted from the open window to obtain the β-dose contribution. The density under the high-Z filter (lead in the example in Figure 17-1-6) gives

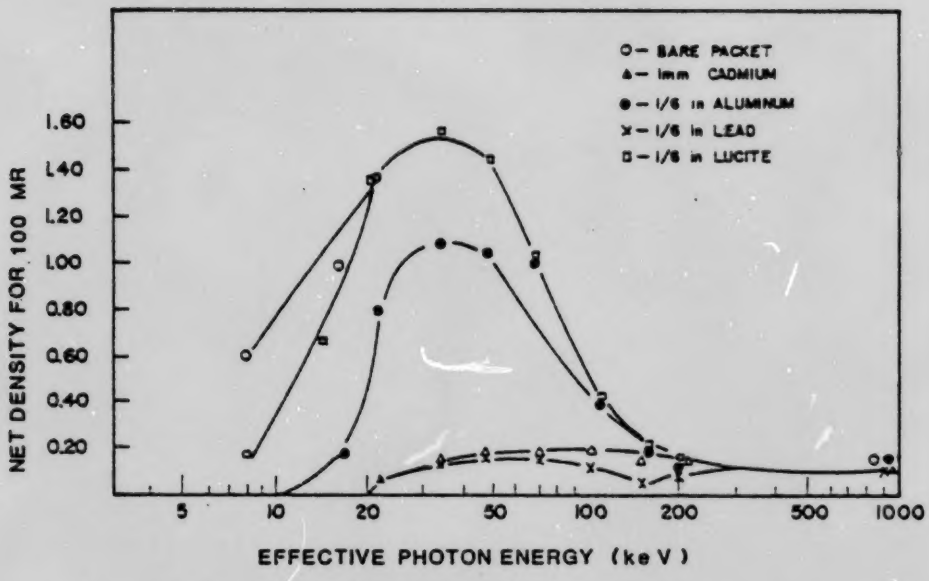

Figure 17-1-6. Photon energy response of film (DuPont 502) in badge through indicated filters.

TABLE 17-1-1

RADIATION PASSED THROUGH VARIOUS FILTERS

FILTER	RADIATION PASSED
High-Z filter	Higher-energy photons
Medium-Z filter	Photons with attenuation of low energies, no betas.
Low-Z	Photons with little or no attenuation at high energies, some attenuation of low energies.
None (open window or bare packet)	β + photons with no attenuation.

the dose from higher energy photons. The ratio of low-Z to medium-Z density readings give an indication of the average energy of the remaining portion of the photon dose; and the proper dose estimate can be made, based on knowledge of the energy dependence of the particular film used.

In actual practice, calibration curves are made at many different energies. Four simultaneous equations are employed in the interpretation of the response characteristics of film, often with the aid of a computer.

Figure 17-1-7 illustrates two film badges in use at different installations. The upper badge, designed for use with a single film packet, clips to the clothing with a simple but effective safety pin. The other is more complex, containing neutron-monitoring film, foils, and radiophotoluminescent glass rods.

Despite the use of badges with filters and sophisticated mathematical techniques to evaluate the results, energy dependence plus the inherent variability in films makes photographic film dosimetry relatively inaccurate, particularly for mixed radiations. For low doses of mixed radiations, errors of ± 50% to ± 200% sometimes occur; however, ± 25% can be achieved in most personnel dosimetry situations in the field with perhaps ± 10% in the laboratory.

To some extent, the errors may be a result of other characteristics of the photographic films. As previously mentioned, conditions of developing must be rigidly controlled; time, temperature, and developer composition can affect the development process. Because films are made in the shape of thin, flat sheets, they are angular dependent, with an edgewise exposure producing a different response per unit exposure than one perpendicular to the plane of the film. Figure

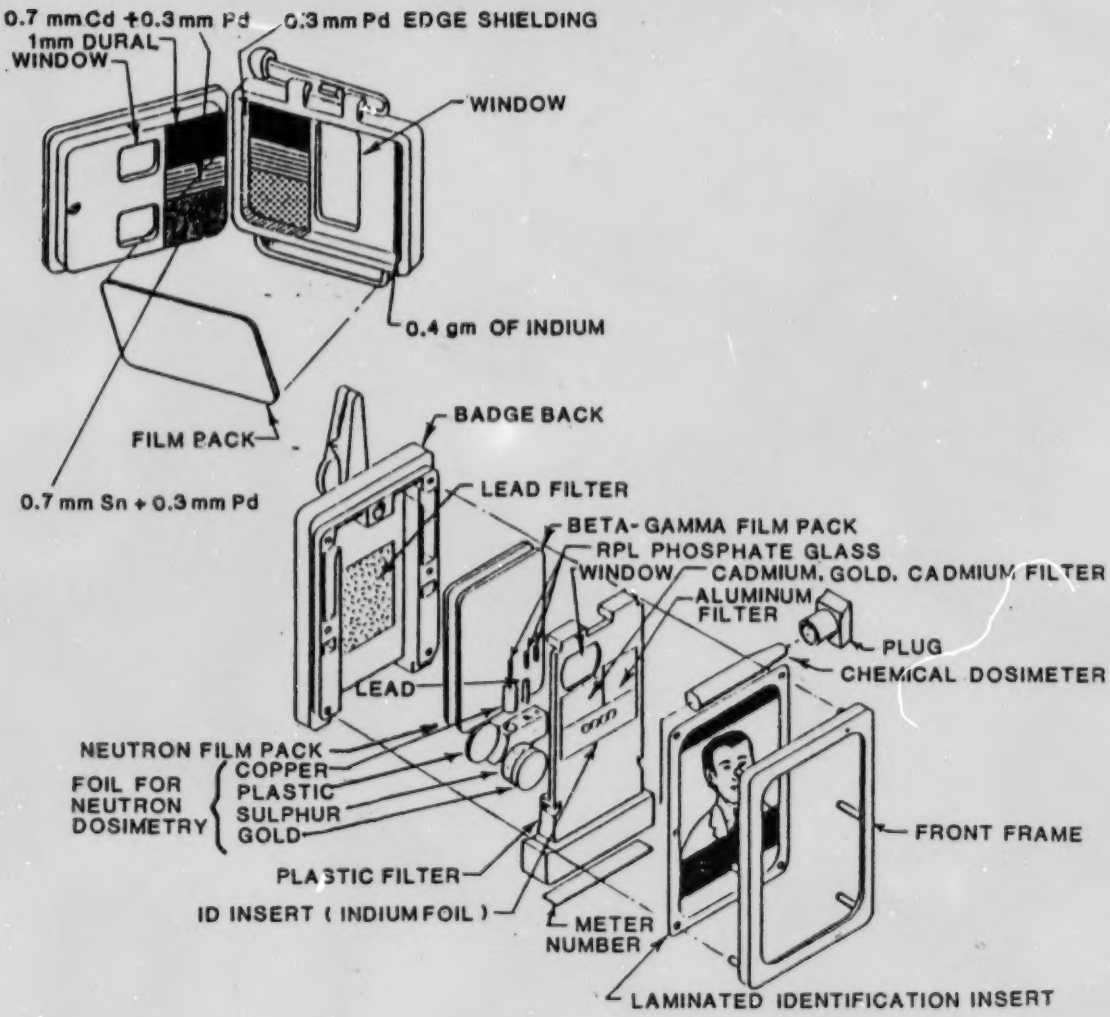

Figure 17-1-7. Two examples of personnel film dosimeters.

17-1-8 depicts the angular dependence of a typical photographic film. For edge-on exposure (i.e. 0^0), the response is much less than for perpendicular exposure. Similarly, the effect is more pronounced with low energies (Curve 1) than with higher energies (Curve 3).

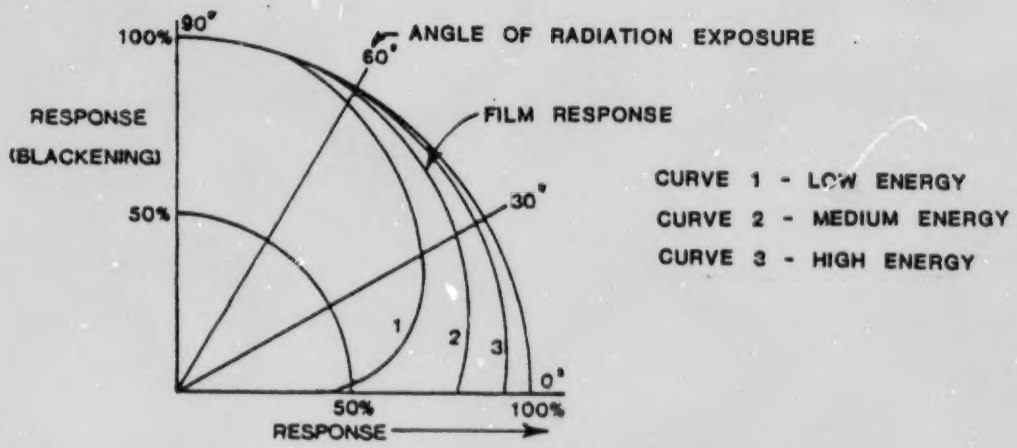

Figure 17-1-8. The blackening of a film vs. the angle of incidence of gamma rays for three different energies.

Films are significantly affected by their environment, especially by temperature and humidity. Environmental conditions may alter the response of the film or affect the stability of the latent image. Under normal conditions, the latent image may last for a few months and then begin to fade. Thus, a film exposed to 1 R may show a density of 1.0 if developed, for example, within a few weeks after exposure. The density produced by the same exposure will be smaller if many weeks pass before the film is developed. Latent-image fading is influenced strongly by the relative humidity, as indicated by Figure 17-1-9. Normally, the higher the humidity, the less stable the latent image; the fading is apparently caused by reoxidation of the silver grains which then may be reincorporated into the matrix of the silver bromide crystal, making them unavailable when the film is developed.

Protection from the effects of humidity can be achieved if the films are sealed in water-tight containers of thin

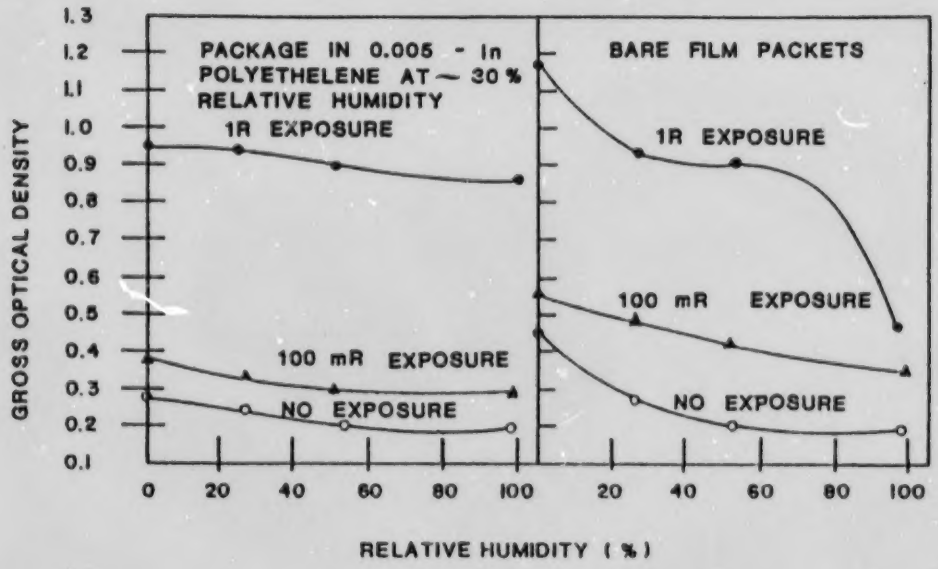

Figure 17-1-9. Gross optical density of personnel monitoring films held at 20°C for 13 weeks after exposure.

plastic. Very high humidity accelerates latent image fading, and even a few days at humidities near 100% will produce some fading. Some films will fog at very low humidities, as also illustrated in Figure 17-1-9; but this is not too great a problem.

The response of photographic films to radiation may be directly affected by temperature, as well as by humidity. At high temperatures, above about 60°C (140°F), fogging or darkening due to heat may occur. Chemicals such as mercury or certain other air pollutants also may diffuse through the film wrapper, altering the response or adversely affecting the latent image. Ozone and similar oxidizing agents can destory or lessen the latent image. Films must be kept in lighttight packages, for they are extremely sensitive to

light; even a pinhole leak may ruin a film. Static electricity may be built up on the surface of the film from handling or rubbing against clothing; nylon clothes are particularly good at generating static charges. When developed, a static discharge "tree" may appear on the film. Other environmental factors may produce effects. Pressure – squeezing a film hard – can cause marks to appear on the developed film, and radioactive contamination can produce a spotty result due to the radiation from contamination spots on the exterior of the film packet.

CALIBRATION AND INTERPRETATION OF FILM

Films are calibrated by being placed in a known radiation field and exposed to a known dose; then the net optical density is read. The net optical density is plotted on a log-log graph along with the known dose to give a calibration curve. In mixed energy fields, the exposure usually is made for several energies in the low-energy region, where energy dependence is predominant. In fields of mixed radiation, exposure is made to several combinations of mixed radiation. From these, a set of curves is developed for each filter, as in Figure 17-1-6. Computers can be programmed to solve the results or to interpret unknown films. The work also can be done manually.

FILM FOR NEUTRON DOSIMETRY

Neutron dosimetry can be accomplished with film. A filter of a material, such as cadmium, with a high thermal neutron cross-section is placed into the film badge. When exposed to thermal neutrons, gamma rays are produced by the (n,r) reaction in cadmium. Cadmium becomes activated and emits beta rays, which darken the film only under that filter. By subtraction of the dose from other causes, the dose from thermal neutrons can be determined.

Ordinarily, however, neutron emulsions are employed for neutron detection. These emulsions are relatively thick, on the order of perhaps 100 micrometers or more. The emulsion is mostly carbon and hydrogen. The proton recoil travels through the film and leaves a track behind (Figure 17-1-10).

Figure 17-1-10. Film, showing protons recoil tracks produced in nuclear track emulsion.

The film is examined under high magnification, and the number of tracks within a given film area is counted. Each view of

the microscpoe is known as a "field." Many fields are counted (20 to 100) to minimize the statistical error. Usually, each proton recoil track is roughly equivalent to a dose of 20 millirem. Some background tracks need to be determined and subtracted from the total.

Nuclear emulsions are highly sensitive to all the problems of ordinary β-γ film, especially to latent image fading, and have other disadvantages, as well. If a fairly high gamma or X-ray exposure occurs along with the neutron exposure, the gammas and X-rays will cause blackening of the emulsion, and the tracks will be difficult or impossible to see. Because the thick emulsions used for high, fast neutrons are sensitive to latent image fading, they cannot be used in high-humidity areas unless sealed in polyethylene or other plastic, and even then can stay out in the field for only a few weeks.

The principle behind the nuclear emulsion is quite simple: fast neutrons, in passing through the hydrogenous (paper) wrappings of the film, knock loose protons that travel through the thick emulsion of the film, leaving behind a track of exposed silver halide grains. At least 450 keV is required to knock loose a proton; thus, neutrons with energies below about 450 keV (actually about 0.8 MeV in practice) will not be detected by the film. The energy dependence of the film is depicted in Figure 17-1-11. Above approximately 4 MeV, the response is fairly good; but below 1 MeV, poor. Below 450 keV, there is no response. Since most reactors have mostly intermediate and thermal energy neutrons in their vicinity, nuclear emulsions may not be a useful dosimeter at reactor sites.

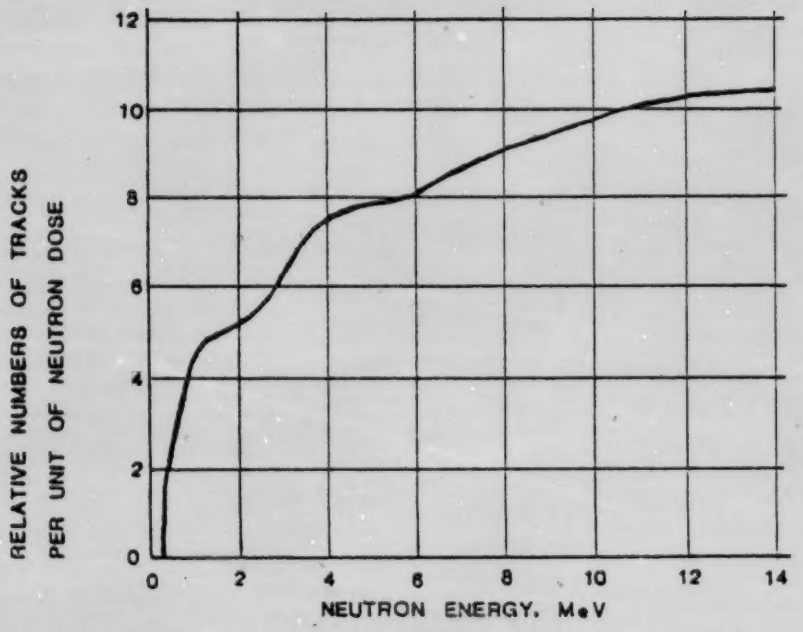

Figure 17-1-11. Energy dependence of a nuclear emulsion for neutron dose.

EXERCISES

1. A film exhibits the following density pattern after being developed:

OW	0.06
Plastic	0.01
Tantalum	0.00

   What would you suspect the exposure to be from?

2. Calculate the probable error for a neutron track emulsion with the following data from 20 fields:

Field	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
Tracks	0	6	3	2	1	0	4	2	5	3	2	1	4	8	1	3	2	5	2	6

   Compare with the data from a background film.

Tracks	0	0	1	0	2	0	4	0	0	0	2	1	0	3	0	1	5	0	0	1

   What is the dose if 1 rad = 1.6 tracks per field?

3. For the film in Exercise 1 above, what fraction or percent of the light is transmitted through the areas under each of the filters? (Hint: Use Equation 17-1-1 or 17-1-3).

MATERIALS

Standard β-γ personnel dosimetry film packets (e.g., Kodak Type 2).
Radiation sources* and calibration.
Darkroom with X-ray safelight.
X-ray developer, stop bath, and fixer.
Developing trays, etc.
Radiation survey meters.
Thermometer.
Densitometer.
Graph paper (log-log, 3 x 2 cycle).
Timer.

*Should have β source (U slab) and X or γ ray source. Neutron source is required (also NTA film) if experiment is to be done with NTA film. Would also need microscope, microscope oil, projecting prism, small projection screen, microscope light, and hand counter.

MAKING A CALIBRATION CURVE

1. Collect a sufficient number of film for a calibration curve. Usually, 3 to 5 film packets should be exposed at each point, and 8-12 points for each curve. Mark each film packet.
2. Position film around source in such a manner that film will receive exposure from 20 mrad to 1 rad. A separate set of control film should be kept in a place at which they will receive no external exposure.
3. With aid of the instructor, place source in position. If an X-ray machine is used, expose each individual film. The films must not be shielded.
4. After exposure, remove film; and develop, along with controls, in appropriate developer. Films should be placed in film developing racks in a known order. If possible, mark the films in the darkroom by making a small code of punched holes. (The films also can be marked with a pencil.)
5. After development, read the film, using a densitometer. Record results, and plot the log of the net optical density against the actual exposure. Note shape of curve, and label all points on it. Use Data Table 17-1-1.

EFFECT OF DEVELOPER TEMPERATURE ON DENSITY

1. Expose 15 films to a convenient dose level; about 100-500 mrad should be adequate.

2. Develop 3 of the exposed films, along with 3 unexposed controls at each of 5 developer temperatures: 65°, 66°, 68°, 70°, and 72°F are suggested.
3. Read the densities on the films, and record in Data Table 17-1-2.
4. Plot the densities obtained as a function of developer temperature.
5. Write a brief explanation of the effect observed.

EFFECT OF DEVELOPMENT TIME ON DENSITY

1. Expose 15 films to a convenient dose level; about 100-500 mrad should be adequate.
2. Develop 3 of the exposed films, along with 3 unexposed films at each of 5 times: 2, 3, 4, 5, and 6 minutes.
3. Read the densities of the developed films, and record in Data Table 17-1-3.
4. Plot the densities obtained for each time on linear paper.
5. Write a brief paragraph describing the effects seen, and give an explanation.

AUTORADIOGRAPHY EXPERIMENT

1. In a small pot, plant several bean seeds. Instructor will water these with water containing ^{32}P. The beans should sprout in about 1 week.
2. One week after the beans have sprouted, pick a leaf and place it atop of a bare film packet. Press it down close to the packet with a book or brick.

3. A few days later (3-5), remove the leaf and develop the film.
4. Explain the results seen. Where is the ^{32}P concentrated in the leaf?
5. Suggest a possible application for autoradiography.

DATA TABLE 17-1-1

Dose	Net Optical Density					Average
	1	2	3	4	5	
0 (control)						

DATA TABLE 17-1-2

Developer Temperature	Density of Exposed Film				Density of Control Film				Average Net O.D.
	1	2	3	Average	1	2	3	Average	
65°F									
66°F									
68°F									
70°F									
72°F									

DATA TABLE 17-1-3

Developer Time	Density of Exposed Film				Density of Control Film				Average Net O.D.
	1	2	3	Average	1	2	3	Average	
2 min									
3 min									
4 min									
5 min									
6 min									

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Dudley, R.A. "Dosimetry with Photographic Emulsions," Ch. 15 in Attix, F.H., Tochilin, E. and Roesch, W. C., eds., Radiation Dosimetry, Vol II, pp. 326-328. New York: Academic Press, 1966.

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Ehrlich, M. "The Use of Film Badges for Personnel Monitoring," IAEA Manual STI-PUB-43 (1962).

Kathren, R.L. and Brodsky, A. "Accuracy and Sensitivity of Film Measurement of Gamma Radiation," Health Physics 9:769 (1963).

Mees, C.E.K. The Theory of the Photographic Process. New York: MacMillian, 1967.

1. Define the following terms:
   a. Latent image.

   b. Optical density.

   c. Solarization.

   d. Sensitivity.

   e. Energy dependence.

   f. Radiophotoluminescent.

   g. Film calibration.

2. Describe the theoretical basis of film dosimetry.

3. What effect does energy dependence have on the interpretation of a beta-gamma film badge?

4. State the effects of two important environmental conditions on the response of a film badge.

5. Nuclear emulsions can be used to measure neutron dose. Describe how this is done.

6. From the procedures performed, describe:
   a. Effect of developing temperature on density.

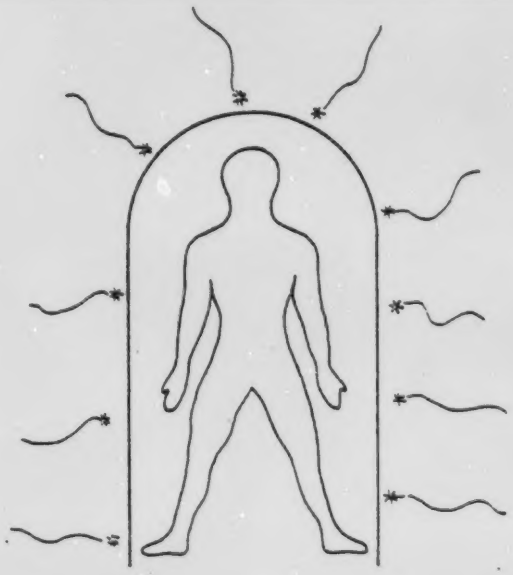
   b. Effect of developing time on density.

   c. An application of autoradiography.

NUCLEAR TECHNOLOGY

MODULE 17 - 2

THERMOLUMINESCENT DOSIMETRY (TLD)



RADIATION PROTECTION II

COURSE 17

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 17 RADIATION PROTECTION II

MODULE 17-1	Film Dosimetry
MODULE 17-2	Thermoluminescent Dosimetry (TLD)
MODULE 17-3	Radiation Dose Assessment - Personnel Dosimeters
MODULE 17-4	Internal Dose Assessment
MODULE 17-5	Bioassay Techniques and Recordkeeping for Radiation Protection

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INTRODUCTION

In the 1950's, more than a half century after the discovery of X-rays and radioactivity, a new type of dosimeter began to be used in certain limited applications for measurement of radiation doses to personnel. This new type of dosimeter was the brainchild of Dr. Farrington Daniels, a physical chemist and professor at the University of Wisconsin. Daniels had applied a phenomenon, known for nearly 300 years, to the measurement of radiation. From its early and very slow start in about 1950, "thermoluminescence," or "TL" as it is simply called, is today the most widely used personnel dosimetry technique in the United States.

What Professor Daniels had rediscovered and refined was the ability of certain chemicals to give off light – visible light – when they are heated. This property is known as "thermoluminescence." The light given off was a measure of how much radiation the chemical had been exposed to. Daniels, an innovative and careful scientist, began to study the effect and the TL process and to determine if it could be made into a practical measurement tool.

Professor Daniels had not been the first to observe thermoluminescence. Indeed, nearly 300 years before, Sir Robert Boyle, a British scientist who also dabbled in chemistry, observed a strange and ghostly light when he heated a diamond in his darkened bedroom in 1663. Others studied the TL phenomenon; and two German scientists, E. Wiedmann and G.C. Schmidt, even suggested its use as a dosimeter for non-ionizing radiations in 1895. Indeed, it was so used in 1935, but not until the late 1940's did work begin in Professor Daniels' laboratory to lead to its use as a personnel dosimeter for ionizing radiation.

The research of Professor Daniels was paid for by the U.S. Army. Other research, this time sponsored by the Navy, began in the mid-150's; and by 1960, TL dosimetry, or TLD, had become a commercial reality.

MODULE PREREQUISITES

The student should have completed Course 1, "Radiation Physics," Course 2, "Radiation Protection I," and Course 11, "Radiation Detection and Measurement," as well as basic courses in chemistry, physics, and algebra.

OBJECTIVES

Upon completion of this module, the student should be able to :

1. Define the following terms:
   - Phosphor.
   - TL Peaks
   - Glow curve.
   - Annealing.
   - TLD reader.
   - Electron traps.
   - Albedo dosimeter.
   - Tribothermoluminescence.
   - Preread anneal.

Forbidden zone.
Valence band.
Conduction band.
Thermoluminescence (TL).
Supralinear.

2. Describe the theoretical basis of thermoluminescent dosimetry.
3. Explain the advantages and limitations of TL for personnel dosimetry.
4. Describe how dose is determined from the glow curve.
5. Draw and label a typical TLD badge or dosimeter.
6. Compare the beta, photon, and neutron response of various TL materials; show how these responses are applied to personnel dosimetry.
7. Given the appropriate equipment –

   Expose a series of chips to a gamma source of known intensity over a range of 4 mR to 4.0 R. Read and record the results. Use glow curves to determine peak heights and integral readout from reader.

   Make 2 calibration curves.

   Anneal all chips according to manufacturer's specifications and redo above.

   With the use of an x-y plotter, record several typical glow curves. Label all peaks.

THEORY OF THERMOLUMINESCENCE

Thermoluminescence, or TL, is a property common to all inorganic crystals and to many other chemicals, as well. In simple words, thermoluminescence means "to give off light when heated." The thermoluminescence process can be shown graphically with the aid of an energy level or solid state diagram. Figure 17-2-1 consists of such a diagram for a typical inorganic crystal. This simplified diagram illustrates three bands, or zones, each describing an energy level for the valence electrons. The lowest energy level for the valence electron is in the valence band, and normally this is where the valence or combining electrons are.

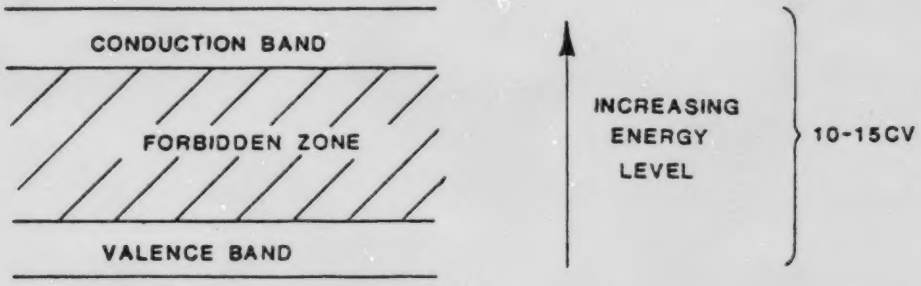

Figure 17-2-1. Simplified schematic energy level diagram.

If, through some outside agency, energy is applied to the system, some of the electrons in the valence band will acquire sufficient energy to move into the conduction band. The electron must acquire a sufficient amount, or quantum, of energy to get into the conduction band, for in a perfect crystal it cannot exist in the energy states between the valence band and conduction band. This range of energy states is

known, therefore, as the "forbidden zone." The phenomenon is illustrated by Figure 17-2-2 and is similar to what occurs in a photographic light meter. When light, an outside source of energy, falls on the sensitive material of the light meter, energy from the light is absorbed by some of the electrons in the valence band, raising them in energy into the conduction band. Here they are free to move about through the crystalline lattice. Moving charges are simply an electrical current, and this is what the light meter measures. When the light is removed, the electrons in the conduction band drop back into the valence band, and the effect (a measurable electrical current flow in the case of the light meter), stops.

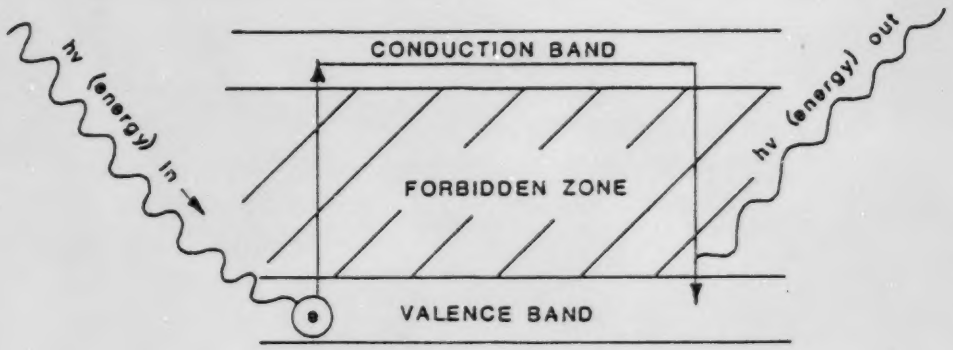

Figure 17-2-2. Schematic representation of what occurs when energy from a photon raises an electron into the conduction band of the crystal.

Note: When the electron drops back into the valence band, it emits the energy it gained, usually in the form of a photon of electromagnetic energy.

A similar process occurs in crystals when they are irradiated with ionizing radiation. The effect described – i.e., electrical current flow – is very much smaller than occurs in

a light meter but, nonetheless, can be measured. This effect, however, is not thermoluminescence: TL is a related, but very different, phenomenon. Most crystals – indeed, virtually all – have various imperfections due to cleavages, points of stresses or strain, or impurities. The imperfections may manifest themselves as electron traps within the forbidden zone. The traps may be actual physical locations within the crystal where tiny electric potentials have accumulated because of a variety of reasons, such as the inclusion of a double positively charged ion (e.g., MN^{++}) in a crystal of monovalent ions such as NA^{+}. As indicated by Figure 17-2-3, the traps can occur at various levels, or depths, within the forbidden zone. Traps, as their name suggests, are places

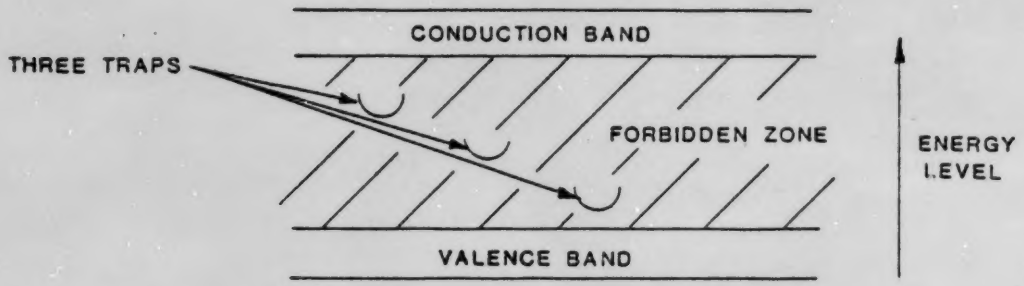

Figure 17-2-3. Energy level diagram showing traps schematically within the forbidden zone.

at which an electron in the crystal may be trapped or held within the forbidden zone, as in Figure 17-2-4.

In Figure 17-2-4, an electron is raised from the valence band into the conduction band by a photon hv_{in}. The electron travels into the conduction band, but when it attempts to return to the valence band, it is trapped at an energy level intermediate between the two bands, emitting a quantum of energy hv_{out} in the process.

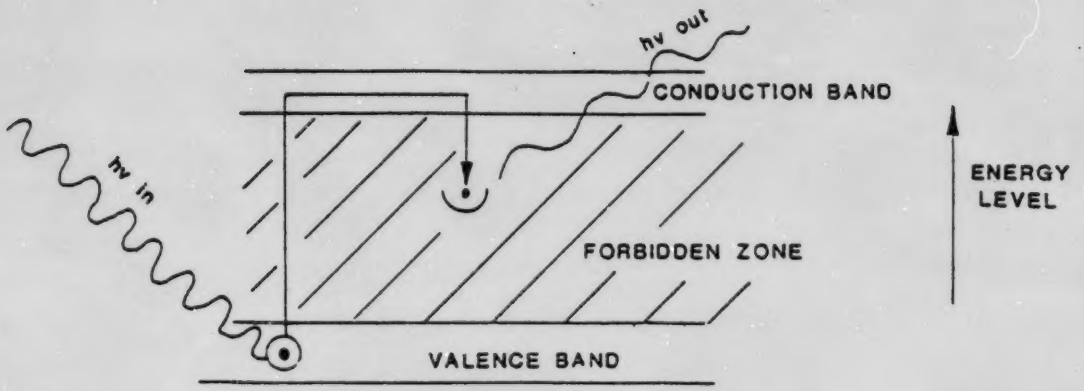

Figure 17-2-4. Energy level diagram illustrating electron trapping.

A trapped electron may exist for years within the conduction band unless something occurs to permit it to escape from the trap. To effect its escape, it requires a small amount of energy, which can be supplied in the form of heat. When heat is applied, as shown schematically in Figure 17-2-5, the trapped electron e is raised once again into the conduction band, migrates through it, and drops back down into the valence band, causing the emission of a photon of visible light hv.

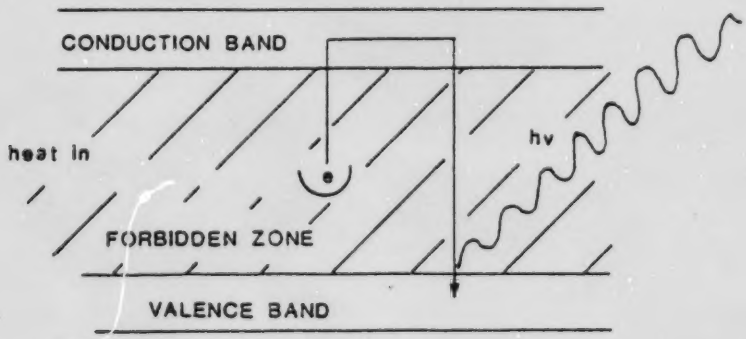

Figure 17-2-5. Schematic representation of thermoluminescence.

The amount of heat required to effect this release depends on how deep the trap is: the deeper the trap, the more heat required. Thus, in a crystal containing three traps, as is depicted in Figure 17-2-3 above, there will be three temperature levels at which electrons will be released from traps. If a plot is made of light emission as a function of temperature, a curve of the type in Figure 17-2-6 will be ob-

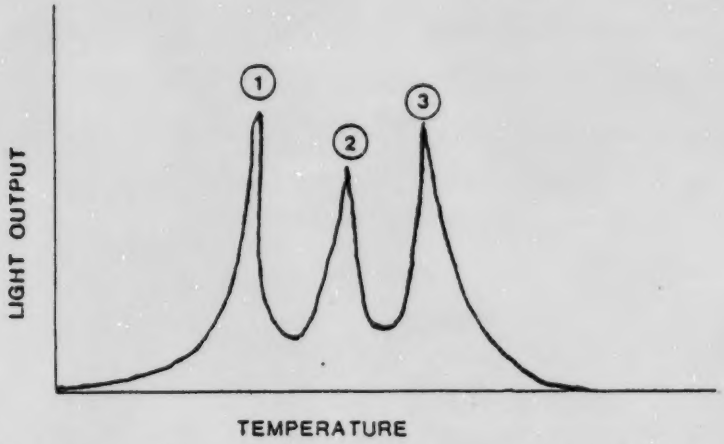

Figure 17-2-6. Schematic glow curve for a TL phosphor having three traps as illustrated by Figure 17-2-3.

tained. This curve is known as a "glow curve." Figure 17-2-7 illustrates an actual glow curve for a typical TL phosphor, LiF, containing magnesium and titanium impurities. Peaks of light output are represented at seven temperatures: -6, 14, 59, 102, 137, 170, and 186°C. Obviously, the two lower temperature peaks are not present at room temperature (20°C), for at room temperature there is enough heat to empty, or "dump," these lower energy traps. Thus, only the higher energy traps are useful for thermoluminescence dosimetry.

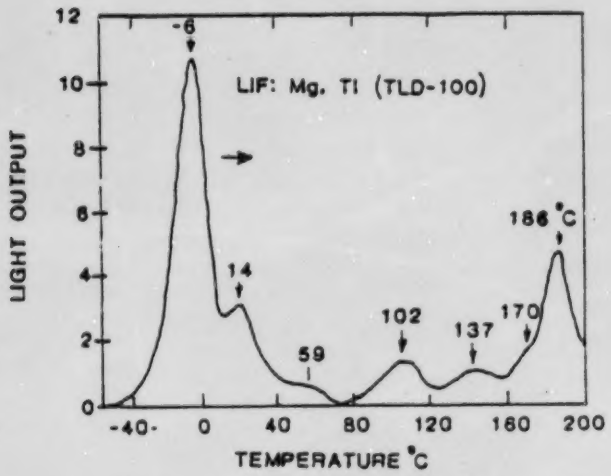

Figure 17-2-7. Glow curve for LiF.Mn.Ti over temperature range of -40 to 200°C.

As a practical consideration, another characteristic of TL needs to be considered. The traps somewhat above room temperature – up to perhaps 100°C or so – are not usable for dosimetry since they are not stable. They fade with time since the release of an electron from a trap follows a probability relationship, P, which is expressed mathematically by Equation 17-2-1.

$$P = Se^{-(E/kT)} \qquad \text{Equation 17-2-1 (Equation = Eq.)}$$

where:

P = the probability of the trap being emptied in a given time.

T = the absolute temperature.

E = the trap depth, or energy level.
k = Boltzmann's constant.
S = a constant related to the atomic vibration frequency of the specific type of crystal.

Thus, TL peaks emitted at temperatures below approximately 150°C are not practically useful. Similarly, peaks emitted at temperatures above approximately 350°C are not usable since either the metal pan on which the TL material is heated, or the TL itself, will incandesce, or glow, thus interfering with the glow-curve measurement.

READOUT AND ANNEALING

Readout of TL materials is relatively simple. Since the amount of light emitted is proportional to the radiation dose to which the phosphor was exposed, only the total amount of light emitted must be measured with a suitable instrument. Figure 17-2-8 illustrates such an instrument, commonly known as a "TLD reader." Alternatively, the dose from a TL dosimeter or TLD can be obtained by measurement of the height of one particular peak. Either method is satisfactory, but the total light output usually is measured, since for any given dose, more light is collected, producing a larger signal. Thus, the total-light method is more sensitive; that is, it allows a given TL phosphor to be used to measure smaller doses. It is also somewhat better from the standpoint of precision and accuracy.

Ordinarily, when a TLD is being read out, certain steps are performed to improve the reading. One very obvious step is the heating of the TL material under very controlled con-

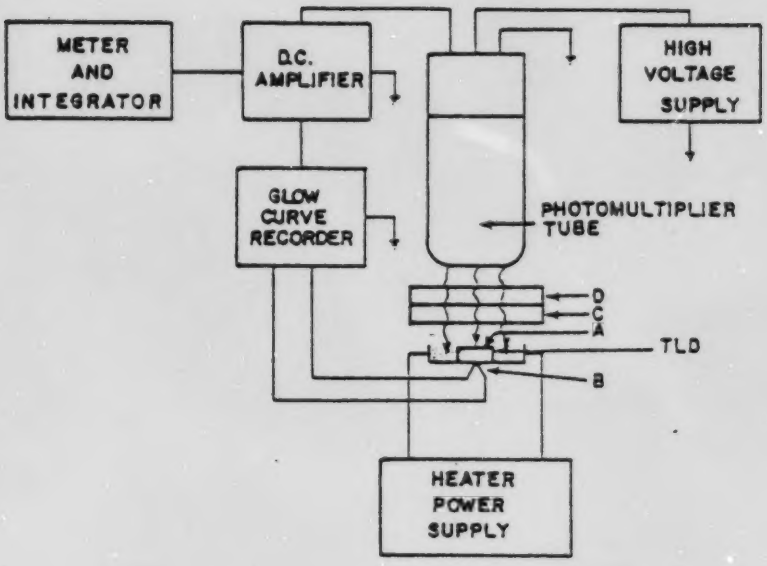

Figure 17-2-8. Schematic diagram of a thermoluminescent dosimeter reader.

ditions. A constant heat rate on the order of 20 to 30°C per second usually is applied to the pan on which the TL is heated. The heating and subsequent readout are done, of course, inside the machine, and care must be taken to prevent any leakage of light into the readout chamber.

The reader also may be programmed to eliminate the reading of low-temperature peaks by starting the heating cycle and allowing the temperature to reach a relatively low level – say 104°C – where it is held for a few seconds. This procedure allows the unstable low-temperature traps to be emptied. No reading is taken at this point. The temperature then is raised again and the reading made. (Refer to Figure 17-2-9).

TL readers really are quite simple in concept. The TL material that has been exposed to radiation is placed on a metal pan in a drawer and closed inside the reader directly

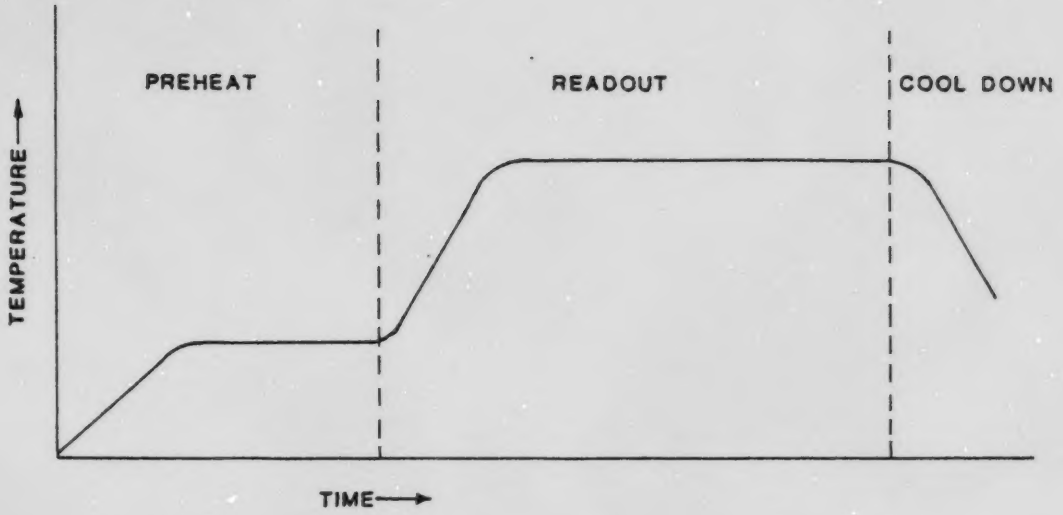

Figure 17-2-9. The preheat or preread methods of eliminating the low-temperature glow peaks.

under a photomultiplier tube. The metal pan – usually silver because of its good heat transfer properties or stainless steel because of its high resistance to tarnish – is electrically heated and, in turn, heats the TL phosphor. At the appropriate temperatures, the TL phosphor emits light, which falls on the photomultiplier cathode, causing electrons to be ejected. These electrons are amplified in the photomultiplier tube, and the resulting current flow is measured by an integrating ammeter that reads out the total charge collected during the readout cycle. Note: Current is simply the total charges per second; hence, the collection of the charges over a period of time gives the total charge, normally expressed in units of nanocoulombs.

Most readers are equipped with a recorder output that provides a voltage signal by passing the current through a resistance of appropriate size. Some readers are designed to read peak height instead of total light output. Peak height

is determined by an electronic circuit that detects the maximum current flow instead of integrating or collecting all the charge during a given time period. There are many types of readers and many TLD's, but the basic readout principles are the same for all.

After readout, the TLD is returned to its original state, according to the theoretical model of thermoluminescence proposed above. Thus, TLD's are reusable over and over again. Some of the electrons, however, are trapped again, particularly in the low-level or energy traps. Thus, after readout, a small fraction – on the order of a few percent or less – of the electrons again populate the traps. If the original dose was sufficiently low, this tiny number of retrapped electrons will have no effect on the next measurement. On the other hand, if the original dose was sufficiently great, the next measurement could be strongly affected.

Consider a situation in which a dose of 1 rad results in a light output of 1000 units on the reader. If 95% of the trapped electrons are returned to the valence band after readout, then a residual of 50 units (equivalent to 50 mrad) will remain. This could introduce a significant error into the next measurement. Moreover, the TLD may not return to its original state by reading it out one, or even several, times; and the sensitivity may change after readout.

These problems can be solved by a process known as "annealing." Annealing is simply a procedure in which the TL material is held for a long time – perhaps minutes or hours – at an elevated temperature to ensure that all the traps are emptied. The specific time and temperature relationships of annealing depend largely upon the TL phosphor and how it is used; however, a typical annealing cycle would be the heating

of the TLD's for 1 hour at 400°C, followed by a 16-hour heating at 80°C. After cooling, the TLD's can be used, read out, and annealed again just before reuse. By waiting to anneal until just before use, any accumulation of background radiation is eliminated. And, by appropriate selection of the annealing process, the phosphors are returned to their original level of sensitivity after each annealing.

TL PHOSPHORS AND THEIR CHARACTERISTICS

There are a great many TL phosphors available, as indicated by Table 17-2-1. The most widely used is LiF, which has excellent sensitivity characteristics, glow peaks in the right temperature range (Figure 17-2-10), and an effective atomic number very close to that of tissue. LiF has the added advantage of being inexpensive. The energy dependence of LiF, therefore, is minimal. Moreover, LiF can be enriched during manufacture with either ^{6}Li or ^{7}Li, which gives the resulting phosphor special properties. ^{7}LiF has essentially no measurable response to neutrons because of the very small neutron cross-sections of ^{7}Li (0.38b) and F (~0.03 b). ^{6}Li, on the other hand, has a very large cross section (942 b) for the n,α reaction with thermal neutrons. Thus, ^{6}LiF is highly sensitive to thermal neutrons as well as to betas and photons, although ^{7}LiF, for all practical purposes, is insensitive to thermal neutrons. In natural LiF, 92.5% of the Li is ^{7}Li with only 7.5% ^{6}Li. Thus, it has a cross section of 71 barns for the n_{th},α reaction, and it is only about 8% as sensitive to thermal neutrons as is ^{6}LiF.

TABLE 17-2-1. SOME COMMERCIALLY AVAILABLE TL PHOSPHORS.

Host crystal	Activators	Trade Name	Manufacturer
BeO	Na, Li		Matsushita, Osaka, Japan
BeO	(Si, Fe, Mg, etc.)	Thermalox 995	Brush Beryll. Co., Elmore, Ohio
BeO	?		Consolidated Beryllium Ltd., Milford Haven, Wales, England
$Li_2B_4O_7$	Mn		Harshaw Chem., Cleveland, Ohio
$Li_2B_4O_7$	Mn, Si		Danish AEC, Risö Roskilde, Denmark
MgB_4O_7	Tb		Dai Nippon Toryo Co., Chigasaki, Japan
LiF	Mg, Ti	"Throw-away"	Radiat. Detect. Co. Mountainview, Calif.
LiF	Mg, Ti	200-Ti	Dohna-lum, VEB Fluowerke Dohna, East Germany
LiF	Mg, Ti		Teledyne-Isotopes Westwood, New Jersey
LiF	Mg, Na	PTL 710	CEC, Montrouge, France
^{7}LiF	Mg, Na	PTL 717	CEC, Montrouge, France
LiF	Mg, Ti	TLD-100	Harshaw Chemical Co., Cleveland, Ohio
^{7}LiF	Mg, Ti	TLD-700	Harshaw Chemical Co., Cleveland, Ohio
^{7}LiF	Mg, Ti	TLD-600	Harshaw Chemical Co., Cleveland, Ohio
Mg_2SiO_4	Tb		Dai Nippon Toryo Co. Chigasaki, Japan
Al_2O_3			CEC, Montrouge, France
CaF_2	Dy	TLD-200	Harshaw Chemical Co., Cleveland, Ohio
CaF_2	Mn		Harshaw Chemical Co., Cleveland, Ohio
$CaSO_4$	Tm		Matsushita, Osaka, Japan
$CaSO_4$	Dy	TLD-400	Harshaw Chemical Co., Cleveland, Ohio
$CaSO_4$	Mn		Harshaw Chemical Co., Cleveland, Ohio

TL phosphors are made according to rigid formulas that provide a certain level of sensitivity and other response characteristics. To the host crystal, certain impurities – most commonly Mn, Mg, Ti, or Dy – are added in controlled amounts, thus producing the proper amount and kind of crystal-

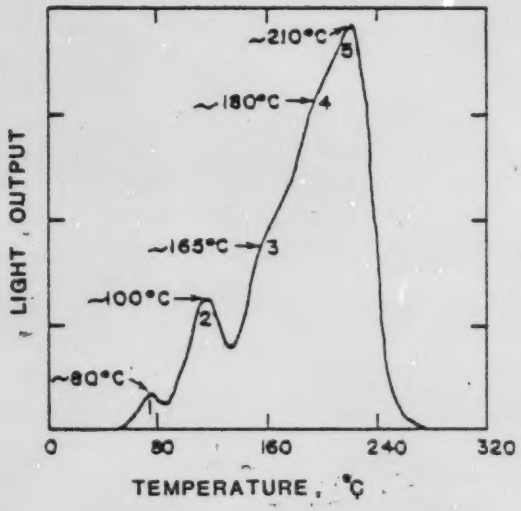

Figure 17-2-10. Typical glow curve for LiF after annealing one hour at 400°C and irradiating 100 R. (Five peaks are shown.)

line lattice imperfections. The phosphor may be "grown" as a large single crystal or may be extruded in the form of long rods. The TL crystal or extrusion then may be cut or cleaved into chips or dosimeters of exact size. A common size used for personnel dosimetry work is a little block that measures 1/8" on a side and 0.04" thick. Another common size is a rod 1 mm in diameter (0.04") and 6 mm (1/4") long in length. Phosphors also may be made in the form of powders, which can be used either as they are or incorporated into other stable matrices such as Teflon plastic.

Since LiF is, by far, the most commonly used dosimeter, the following discussion of TL phosphor characteristics will focus on it. LiF is useful over a wide range of doses to x or gamma rays. A typical LiF chip will have a usable range from, perhaps, as low as 10 mrad to as high as 10^5 rad. This

is an extraordinary range, spanning fully 7 orders of magnitude; therefore, LiF can be used employed not only as a personnel monitor but also for measurement of high-level doses. TL output is linear with dose up to about 700 R. Above this level, the response becomes supralinear (Figure 17-2-11); that is, for each additional dose increment, there is greater than 1:1 increase in light output. The cause of supralinearity is not known, and it does not occur in other phosphors (Figure 17-2-11). The effect is not serious, however, and in no way adversely affects the use of LiF as a dosimeter.

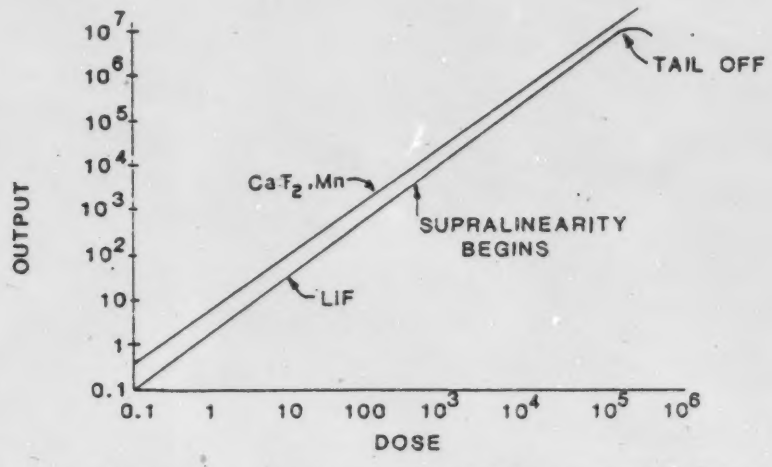

Figure 17-2-11. Typical response curves for LiF and CaF_2, Mn dosimeters.

The lower level of sensitivity of TL dosimeters is determined largely by the sensitivity, or amount of light emitted per unit exposure. The sensitivity can be varied to a great extent by the amount and type of impurities or dopants introduced into the cyrstalline matrix, and it is also affected by the presence of spurious effects. Some of the spurious effects or background can be eliminated by readout in an inert

gas, such as dry nitrogen, rather than air. Readout extends the lower limit of sensitivity and improves precision.

Spurious response or signals may be a result of tribothermoluminescence, which is simply thermoluminescence introduced as a result of mechanical energy or frictional effects absorbed by the TLD. These are largely eliminated by readout in N_2 and by careful handling to avoid shock. Other spurious thermoluminescence can be introduced by dirt or grease on the TLD or heater pan. For this reason, TLD chips or TLD-teflon usually are cleaned in a bath of ethyl alcohol prior to readout.

The upper limit of 10^5 rad for LiF is dictated by radiation damage to the phosphor. Other phosphors have a somewhat higher upper limit but show similar radiation damage that cannot be annealed out. Thus, a typical LiF dosimeter is good for a total dose of 10^5 rad. At personnel exposure levels on the order of 1 rad or less, an LIF TLD is theoretically good for more than 100,000 uses. In fact, TLD's have been used over and over for several hundred times.

TLD's typically exhibit very good precision and accuracy of measurement. Under laboratory conditions, a large group of TLD's exposed to the same dose will exhibit variations of only a few percent or less. Precision usually is with $\pm$5%, and a careful worker normally can obtain values of within $\pm$10 to 15% of the true value.

Many TLD's possess remarkably good energy dependence characterisitcs. Unlike film whose sensitive components (silver and bromine) have atomic numbers much greater than tissue, TL materials such as LiF and $L_2B_4O_7$ have an effective atomic number roughly equal to that of soft tissue. Hence, they respond or absorb energy from photon radiations much like soft

tissue does. Other TL materials, such as CaF_2 or $CaSO_4$, contain materials of higher atomic number and, thus, have a significant overresponse to low energies (Figure 17-2-12). If these are to be used for personnel dosimetry, a badge that contains filters, much like a film badge, will be necessary. For the lower-Z phosphors, such as LiF, filters may not be required since the error produced by energy dependence is so small.

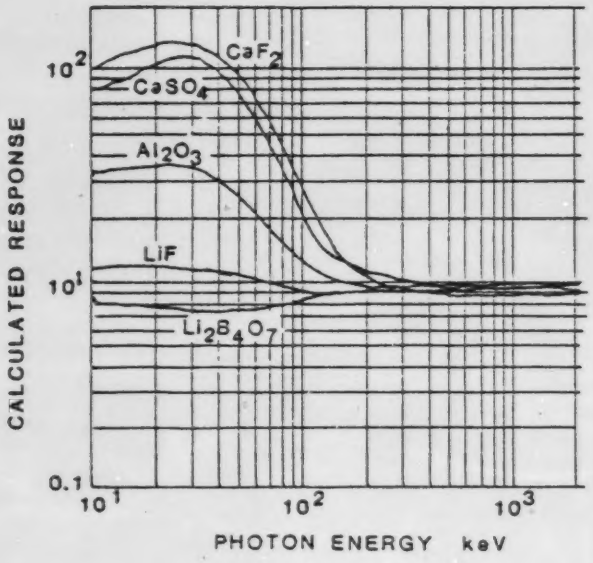

Figure 17-2-12. Calculated photon energy dependence for five different TL materials.

The TL materials also are sensitive to beta radiation, but usually only about half as sensitive as they are to photons. A relatively satisfactory TLD badge for photon and beta monitoring can be made from two LiF or other low-Z chips, – one bare, or "open window," and the other behind a filter to remove, or screen out, the betas. Thus, the bare, or open-window, TLD give the combined dose from beta and pho-

ton radiations; and the shielded TLD gives the dose from photons only. The beta dose is obtained simply by subtracting the shielded TLD reading from that of the bare TLD and interpreting the residual with the aid of a beta dose calibration curve.

A great advantage of TLD's lies in their resistance to the environment. Unlike film, neither the response nor the readout of TLD's is affected by temperatures and humidities normally encountered in the human working environment. Temperatures on a car dashboard in the southwest will reach ~200°F and do have an annealing effect upon LiF TLD chips. Thus, TLD's can be worn over a temperature range from below zero to 104°F and in very high- and low-humidity situations. They also can be worn for very long intervals or, in the case of those utilized for environmental monitoring, left in the field for periods of a year or more. There is no loss of data from long wearing intervals.

Most TL materials do undergo a "fading" effect similar to the latent image fading of photographic films. The mechanism is, however, very much different; and, unlike film fading, TLD fading poses no particular problems. Fading in TL materials normally is a rapid, relatively small phenomenon that occurs because of the emptying of low-temperature traps. The probability of any given trap being emptied is a function of temperature and trap depth, as evident from Equation 17-2-1. (Refer back to this equation.) Because the term E is a measure of trap depth, at a constant temperature (in this case, room temperature), smaller trap depths (the low-temperature traps) have a greater probability of emptying since as E gets smaller, the term $e^{-E/kt}$ gets larger and, of course,

so does P. Thus, a TL readout immediately after exposure will give a higher light output than one that has been inactive for a few days.

In some cases, the light output will stabilize a few days after exposure if the TLD is held at room temperature. In other words, the TLD readout will decrease rapidly at first and then level off, as indicated by Figure 17-2-13.

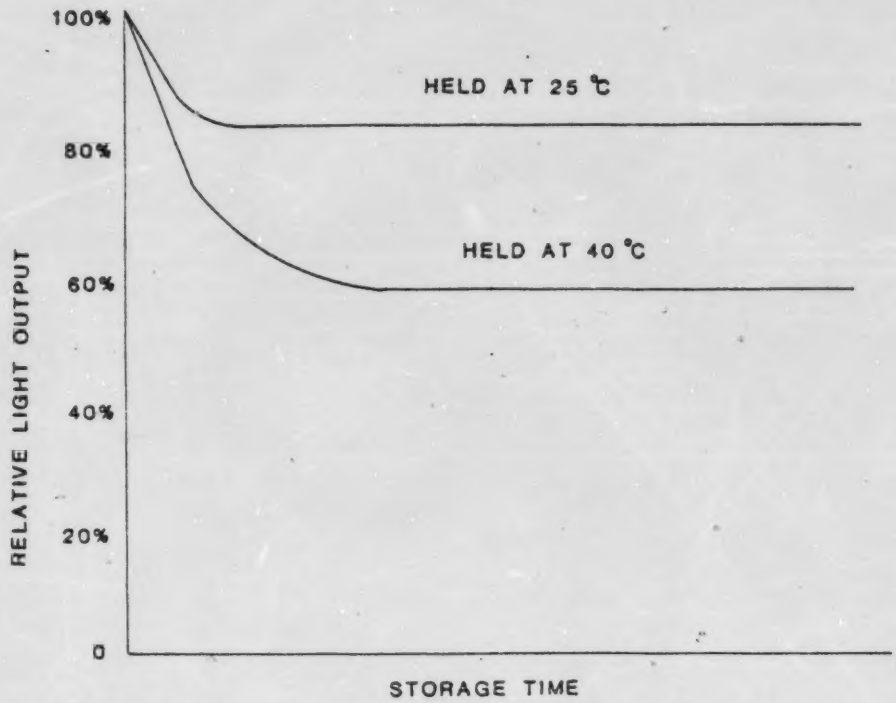

Figure 17-2-13. Fading characteristics of a TL phosphor.

Thus, the problem can be eliminated in some cases merely by waiting a few days before reading out the TLD. A better and more applicable solution is to use the pre-readout technique described above to empty the low-temperature traps. The light from these will not be included in the readout; the readout will include only the light from the higher-temperature traps.

Another way to avoid the problem of fading caused by spontaneous emptying of the low-temperature traps is to preread anneal the TLD's at a fairly low temperature for a few minutes prior to readout. This is a common method of eliminating fading errors.

APPLICATIONS OF TL TO DOSIMETRY

The applications of TL to personnel dosimetry for routine β-photon personnel dosimetry have been mentioned already. Many badges that contain one or two TLD chips have been developed. If LiF is used, only a single chip is needed for photon dosimetry, regardless of energy. For mixed β-photon dosimetry, two chips are required.

TLD's can be utilized for neutron dosimetry, but this task is quite difficult and usually not too accurate. In mixed neutron-photon or neutron-photon + beta fields, ^{7}LiF can be employed to obtain a measurement that is free of neutron response. Natural LiF or ^{6}LiF can be employed in the measurement of the photon and beta response plus the response from thermal neutrons. Thermal neutrons interact with ^{6}Li according to the reaction –

$$^{6}_{3}\mathrm{Li} + {}^{1}_{0}\mathrm{n} \longrightarrow {}^{3}_{1}\mathrm{H} + {}^{4}_{2}\mathrm{He} + Q.$$

The alpha particle and triton deposit all their energy in the TLD since their range is so small; and this, of course, produces a large amount of light per rad of exposure since

the thermal neutron cross section for this reaction is very large (948 barns). This method cannot be employed for neutron dosimetry, however, because the fast neutron cross section is very low.

Fast neutrons are measured by means of an albedo dosimeter. "Albedo" refers to scattered neutrons, and the albedo dosimeter measures neutrons scattered from the body into the dosimeter. These are much reduced in energy and, hence, may be thermalized. In any case, the cross section increases with decreasing neutron energy; therefore, the dosimeter will have a greater response to these albedo neutrons. Unfortunately, the cross section is strongly dependent upon neutron energy, and the dosimeter responds to thermal neutrons and others, in addition to the albedo neutrons. Thus, the light output per rad of neutron exposure is highly dependent upon the neutron energy spectrum; and unless this is known and duplicated by the calibration procedure, an albedo dosimeter will provide little more than a guess or gross indication of the dose. The problem is complicated further when the neutron dose equivalent is determined by the quality factor.

TLD's also have some interesting applications other than personnel dosimetry. They are, of course, highly useful as environmental monitoring devices. Because of their small size, they actually can be implanted inside tissues of experimental animals to measure the dose received. A most interesting application of the TL principle deals with measurement of the the self-irradiation of rocks from the tiny amounts of uranium or thorium in them. The amount of thermoluminescence of undisturbed rocks is a measure of their radiation dose over a period of thousands of years. If the uranium and thorium content are known (say, by X-ray analysis), the age of

the rock can be calculated from the dose determined from the thermoluminescence measurement.

EXERCISES

1. Given Equation 17-2-1 and S = 0.1 per day, calculate the probability of emptying a trap with a depth E equivalent to 80°C in 1 day and 1 week at 25°C, 50°C, and 80°C.
2. A TLD is calibrated to give 1 reader unit per rad for gamma radiation and 0.5 reader unit per rad of beta radiation. Determine the dose to a badge whose reading was as follows:

   Open window = 0.146 reader units.
   Filtered TLD = 0.101 reader units.

MATERIALS

Procedure 1

TLD powder.
Hot plate.
Aluminum broiling pan.

Procedure 2

TLD phosphors (LiF, CaF_2 [or $CaSO_4$, Dy]).
Appropriate exposure source(s).
TLD reader.
Graph paper (log-log 3x3 cycles).
Recorder for glow curve.
Ruler (with which to measure peak heights).

PROCEDURES

Procedure 1

1. The instructor will expose the TLD powder to a dose of about 100 rad.
2. Allow 10 minutes for eyes to adjust to the dark.
3. In a darkened room the pan that contains the exposed TLD should be placed on the hot plate, which is allowed to heat up. Watch for the flash of light from the TLD. BE CAREFUL! DO NOT TOUCH THE HOT PAN OR PLATE.

Procedure 2

1. Expose 2-4 TLD's of each type to at least 8 doses over the range of mrad to 4 rad. The following are suggested: 0, 4, 10, 40, 400, 1000, 2000, and 4000 millirad.
2. Allow the TLD's to sit overnight.

3. Read out each of the TLD's, recording the number opposite the dose in Data Table 17-2-1. Also, obtain glow curves.
4. On 3x3 cycle log-log paper, plot the light output (y axis) vs. the dose (x axis).
5. Calculate and record the light output per rad for each phosphor.
6. Determine the practical minimum sensitivity level for each phosphor from the curves obtained.
7. Measure and record the height of the largest peak for each phosphor at each dose level. Plot this on the 3x3 cycle log-log paper vs. dose.
8. Label on Figure 17-2-14 a typical glow curve for each phosphor.
9. Anneal half of the TLD's according to the manufacturer's instructions and repeat Steps 1-4 above. Glow curves need not be obtained.
10. Repeat for the group that was not annealed. Note differences in response characteristics. What caused these differences?
11. Calculate the light output per rad for the dosimeters that were annealed. How does it differ from the results obtained before? Why?

DATA TABLES

DATA TABLE 17-2-1.

TLD Type _____		
Dose	Light Output	Peak Height

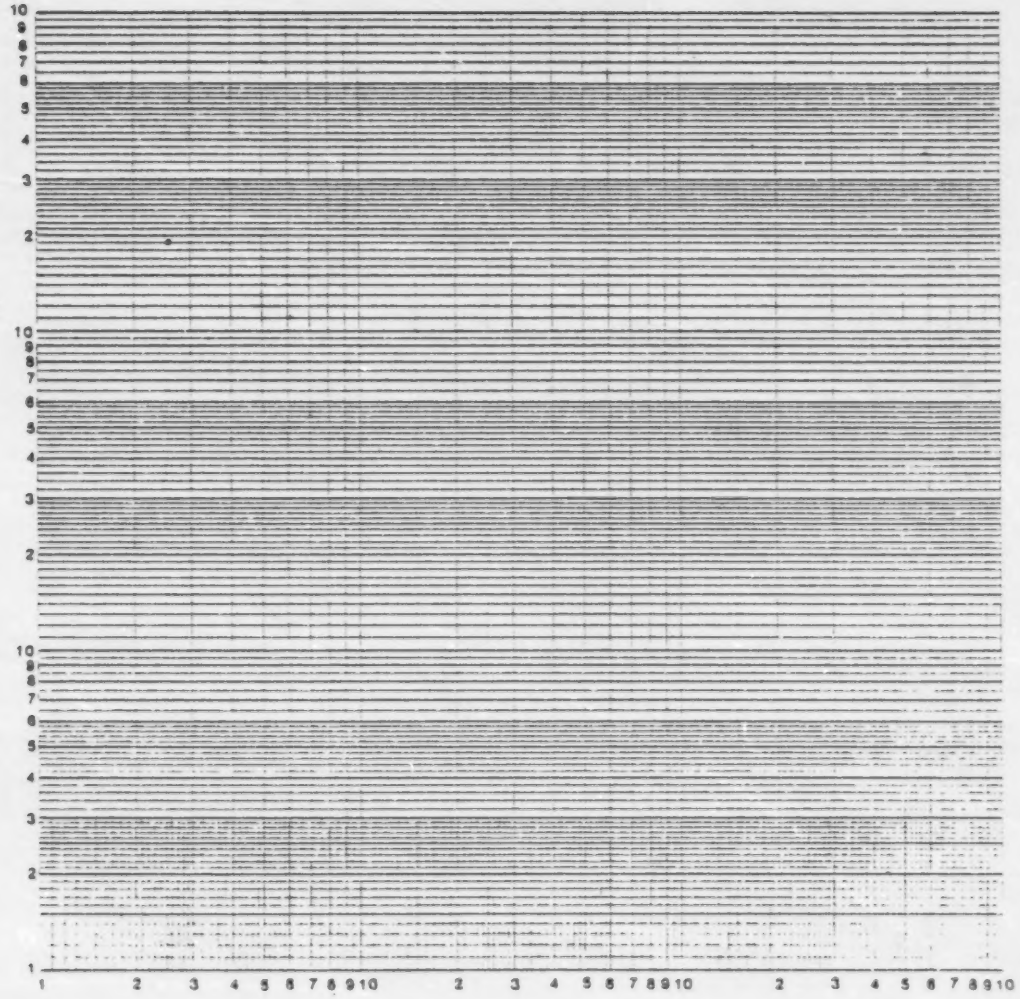

Figure 17-2-14. Glow-curve form.

REFERENCE MATERIALS

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Becker, K. Solid State Dosimetry. Cleveland, OH: CRC Press, 1973.

Cameron, J.R.; Suntharalingham, N.; and Kenney, G.N. Thermoluminescent Dosimetry. Milwaukee, WI: University of Wisconsin Press, 1968.

Cameron, J.R., et al. "Thermoluminescent Radiation Dosimetry Utilizing Lithium Fluoride," Health Physics 10:25 (1964).

Cox, F.M. "New Solid Lithium Fluoride Thermoluminescent Dosimeters," and "New Solid $CaSo_4$, CaF_2, and $Li_1B_4O_7$: Hn TLD's." Solon, OH: Harshaw Chemical Co.

Proceedings of the Second International Conference on Luminescence Dosimetry. CONF-680920 (1968).

1. Define the following terms:
   a. Phosphor.

   b. TL peaks.

   c. Glow curve.

   d. Annealing.

   e. TLD reader.

   f. Electron traps.

   g. Albedo dosimeter.

   h. Tribothermoluminescence.

   i. Preread anneal.

j. Forbidden zone.

k. Valence band.

l. Conduction band.

m. Thermoluminescence (TL).

n. Supralinear.

2. Describe the theoretical basis of thermoluminescent dosimetry.

3. List and describe the advantages and disadvantages of TL when used as personnel dosimetry.

4. Describe how the dose may be determined from a glow curve.

5. Sketch a typical TLD badge or dosimeter, and label each part.

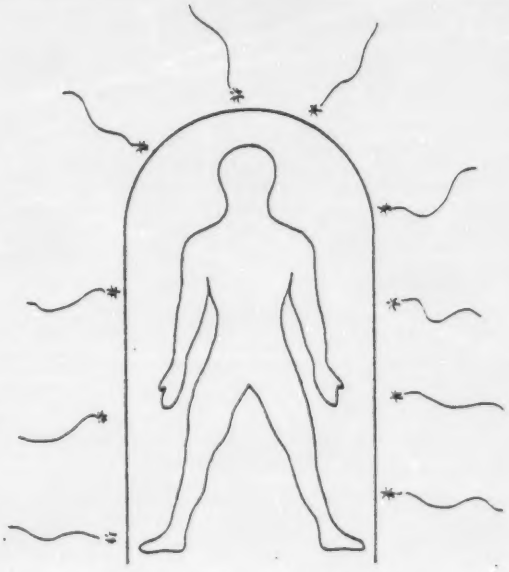
6. Make a comparison of the beta, photon, and neutron response of various TL materials.

7. Explain how the answers to question 6 above are applied to personnel dosimetry.

NUCLEAR TECHNOLOGY

MODULE 17 - 3

RADIATION DOSE ASSESSMENT - PERSONNEL DOSIMETERS



RADIATION PROTECTION II

COURSE 17

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 17 RADIATION PROTECTION II

MODULE 17-1	Film Dosimetry
MODULE 17-2	Thermoluminescent Dosimetry (TLD)
MODULE 17-3	Radiation Dose Assessment - Personnel Dosimeters
MODULE 17-4	Internal Dose Assessment
MODULE 17-5	Bioassay Techniques and Recordkeeping for Radiation Protection

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INTRODUCTION

The previous two modules in this course – "Film Dosimetry" and "Thermoluminescent Dosimetry" – discussed the two materials most frequently utilized in the measurement of external doses that might be received by radiation workers. External doses are those that arise from sources of radiation outside the body. Thus, external doses may be produced by a variety of sources, including beta- and gamma-emitting radionuclides, X-ray machines, particle accelerators, and sources of neutrons that include nuclear reactors. Neither film nor luminescent dosimeters respond to radiations from all these sources in the same way that people do. And since radiation dose to people is the concern, special applications of film and thermoluminescent materials have been developed to determine the doses that radiation workers receive.

Unfortunately, neither film nor thermoluminescent dosimeters can measure external dose from all the radiations to which people can be exposed. Thus, other dosimeters also may be used. This module discusses some of the other dosimeters that are used, including radiophotoluminescence detectors and neutron activation analysis. Since film and thermoluminescent dosimeters were covered in Modules 17-1 and 17-2, this module does not focus on the problems of dose interpretation from a given type of dosimeter or badge, but discusses special-purpose dosimeters such as finger rings.

The student should know the meaning of several other terms before beginning the Subject Matter of this module. "Personnel dosimeter" is an important term; it refers to a device worn by people for the purpose of determining their radiation dose. The word "dosimeter" has two meanings: (1) a device used to measure radiation dose, and (2) a material

(e.g., film or TLD) that responds to (and, therefore, is used to measure) the radiation. A film badge, which consists of one or more films in a holder that contains filters especially designed to correct for the energy dependence of the film, is a dosimeter, but so is the film itself. To avoid confusion, care must be taken in using the word "dosimeter." One way to avoid confusion is to use the term "personnel dosimeter" in place of the single word "dosimeter" when describing a badge or "finished product."

MODULE PREREQUISITES

The student should have completed courses in basic chemistry and physics; Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 11, "Radiation Detection and Measurement"; and the first two modules of Course 17.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe the theoretical basis of radiophotoluminescence dosimetry.
2. Describe and identify applications for finger and wrist dosimeters.
3. Describe the applications and limitations of typical high-range personnel dosimeters.

4. Identify fission and activation foils commonly used in personnel dosimetry; write the nuclear equations for their reactions.
5. Calculate the activity from an activated foil.
6. Define the following terms:
   - Fission foil.
   - Fission fragment.
   - RPL.
   - Track etching.
   - NAD.
   - Personnel dosimeter.
   - External dose.
   - Penetrating dose.
   - Nonpenetrating dose.
7. Describe spark-counting and track-etch detectors.

RADIOPHOTOLUMINESCENT GLASS DOSIMETRY

Radiophotoluminescent glasses are utilized routinely in some countries (e.g., Japan, Germany) for personnel dosimetry, and have other applications as well. In the United States, these glasses are not commonly used for personnel monitoring, except for high-range or "accident" gamma dosimeters. Other applications of these glasses are in research, where a tiny glass rod can be implanted inside a living animal in order that actual in-place measurements of dose can be made.

The basic principle of radiophotoluminescence is simple. Material that is originally nonluminescent when exposed to visible or ultraviolet light undergoes a change when exposed to ionizing radiation. Then, after irradiation, the material will luminesce, or give off visible light when under ultraviolet light. Just how this phenomenon occurs is not completely understood. What is known is that glasses must contain silver for the effect to take place, and the silver usually is placed in the glass in the form of the metaphosphate, which has the chemical formula $AgPO_3$. Thus, these glasses sometimes are known as silver metaphosphate glasses. There are many different formulations or compositions of silver metaphosphate radiophotoluminescent glasses. Some of the more common ones are listed in Table 17-3-1. All of the glasses contain aluminum in addition to Ag, P, and O; many contain Li as well.

TABLE 17-3-1.

ELEMENTAL CONTENT OF COMMON COMMERCIAL RPL GLASSES.

Manufacturer	Type	Weight Percentage of							Relative Sensitivity	Relative Background
		Ag	Al	Li	Na	B	P	O		
Toshiba, Japan	FD-1	3.7	4.6	3.6	--	0.85	33.5	53.7	1.16	2
	FD-3	3.3	5.1	3.6	--	--	34.5	53.5	1.16	2
	FD-5	0.52	6.1	--	8.9	--	33.1	51.3	1.68	1
	FD-7	0.17	6.1	--	11.0	--	31.5	51.2	1.89	≤ 1
Jenaer Glaswerke, Germany	RPL-I	4.6	3.1	4.45	--	--		52.8	1.11	3
	RPL-II	3.67	4.67	3.47	--	0.9		57.7	1.11	3
	RPL-III	0.62	0.51	7.34	4.4	0.95		55.9	1.63	5
	RPL-V	0.58	6.1	--	8.86	--		50.2	1.26	3
CEC, France	Type 1	2.9	3.5	2.5	4.7	--	33.6	52.3	1.00	6
	Type 31	2.4	3.5	0.7	4.8	--	34.1	52.8	1.00	6

A typical glass is manufactured from a base or glass matrix of aluminum phosphate [$Al(PO_3)_3$] mixed with one or more other metallic metaphosphates such as magnesium metaphosphate [$Mg(PO_3)_2$], barium metaphosphate [$BA(PO_3)_2$], sodium metaphosphate [$NaPO_3$], potassium metaphosphate [KPO_3], or frequently the metaphosphate of lithium [$LiPO_3$]. Sometimes lithium oxide/boron oxide glasses are made. A small amount of $AgPO_3$ is added to the glass in order to produce a clear durable glass that exhibits the radiophotoluminescence (RPL) phenomenon. The glass may be thought of as a solid solution of $AgPO_3$ in the glass matrix.

The way in which RPL glasses are thought to work is similar, in many respects, to the mechanism of thermoluminescence. Irradiation of the glass raises electrons into the conduction band of the crystalline glass matrix. When the electrons fall back down to the lower-energy valence band,

some of them are trapped by positively-charged silver atoms, producing a stable fluorescing center with absorption bands in the near ultraviolet. These centers, therefore, will be activated by ultraviolet light. When they are activated, energy transitions occur, and a photon of lower-energy visible light is emitted. The fluorescing center is not destroyed or changed in the process; it can be activated again and again. Indeed it is activated continuously as long as the ultraviolet light illuminates the glass. The process is shown diagrammatically in Figure 17-3-1.

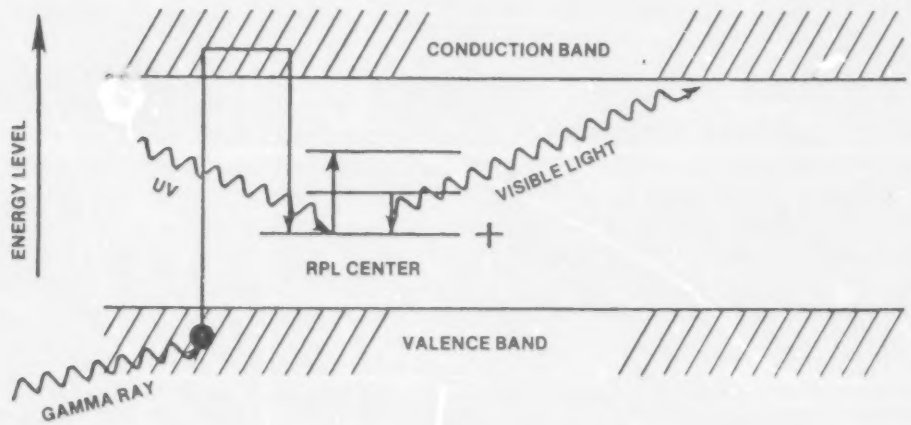


Figure 17-3-1. RPL glass.

In this simplified drawing, an electron, represented by the black dot, is raised from the valence band to the conduction band by a gamma ray. Instead of returning to the valence band, the electron is trapped in a stable RPL center, which has a net positive charge. A photon of ultraviolet (UV) light then can be absorbed by the RPL center, which reemits the UV as a photon of visible light.

Because RPL glasses contain silver (Z = 48) and perhaps other high-Z elements, they exhibit energy dependence for X and gamma radiation, much as does film. Glasses, however, are composed mostly of elements closer in atomic number to tissue; hence, they are not as energy dependent as film. The smaller the amount of silver, the better the energy dependence, as indicated by Figure 17-3-2. In general, however, the smaller the amount of silver, the less sensitive the glass to gamma radiation. Thus, there is a tradeoff between sensitivity and energy response.

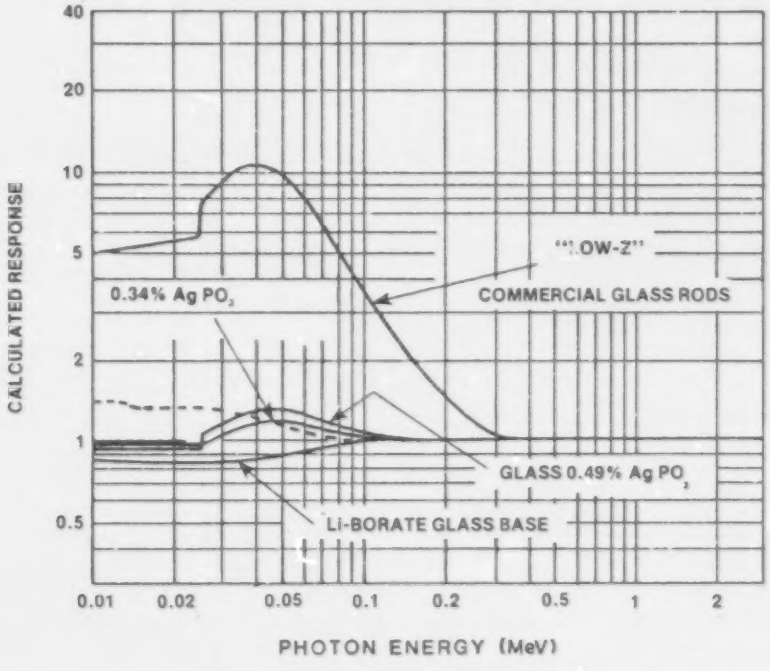


Figure 17-3-2. Energy dependence of several RPL glasses. (Notice that without the inclusion of silver, there is no overresponse at low energies.)

RPL glasses are manufactured in standard sizes for use as dosimeters. The amount of RPL or light emitted is directly related to the size of the dosimeter. The more glass, the more RPL centers and, hence, the more light emitted following irradiation. Glass blocks measuring 8 mm on a side and 4.7 mm in thickness are employed in Europe and Japan for routine personnel dosimetry. Such blocks have a useful range of from perhaps 20 mR to a few hundred R. They typically are enclosed in special shields (Figure 17-3-3) that minimize the energy dependence. The shields are provided with small holes in order that there will be some penetration of beta and low-energy photon radiation.

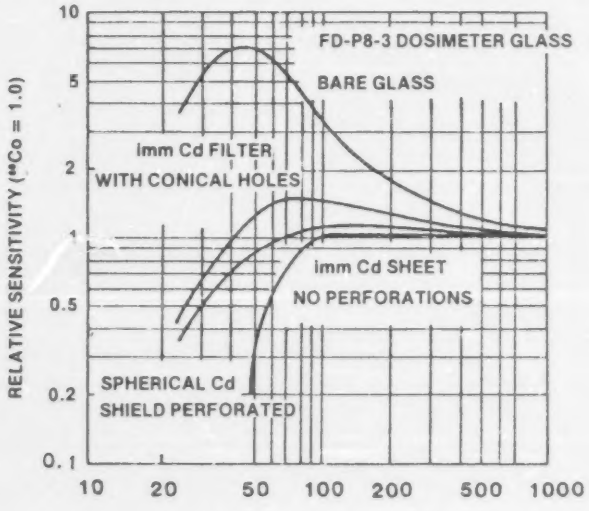


Figure 17-3-3. Photon energy dependence of response of Toshiba FD-P8-3 disometer glass blocks, bare and with different types of perforated Cd filters.

In the United States, a small rod, called a fluororod, of RPL glass is used as a high-range dosimeter, usually in conjunction with a film of TLD. A typical dosimeter that incorporates fluororods is depicted in Figure 17-3-4. The badge illustrated contains two fluororods: one is bare and the other is an energy-flattening filter.

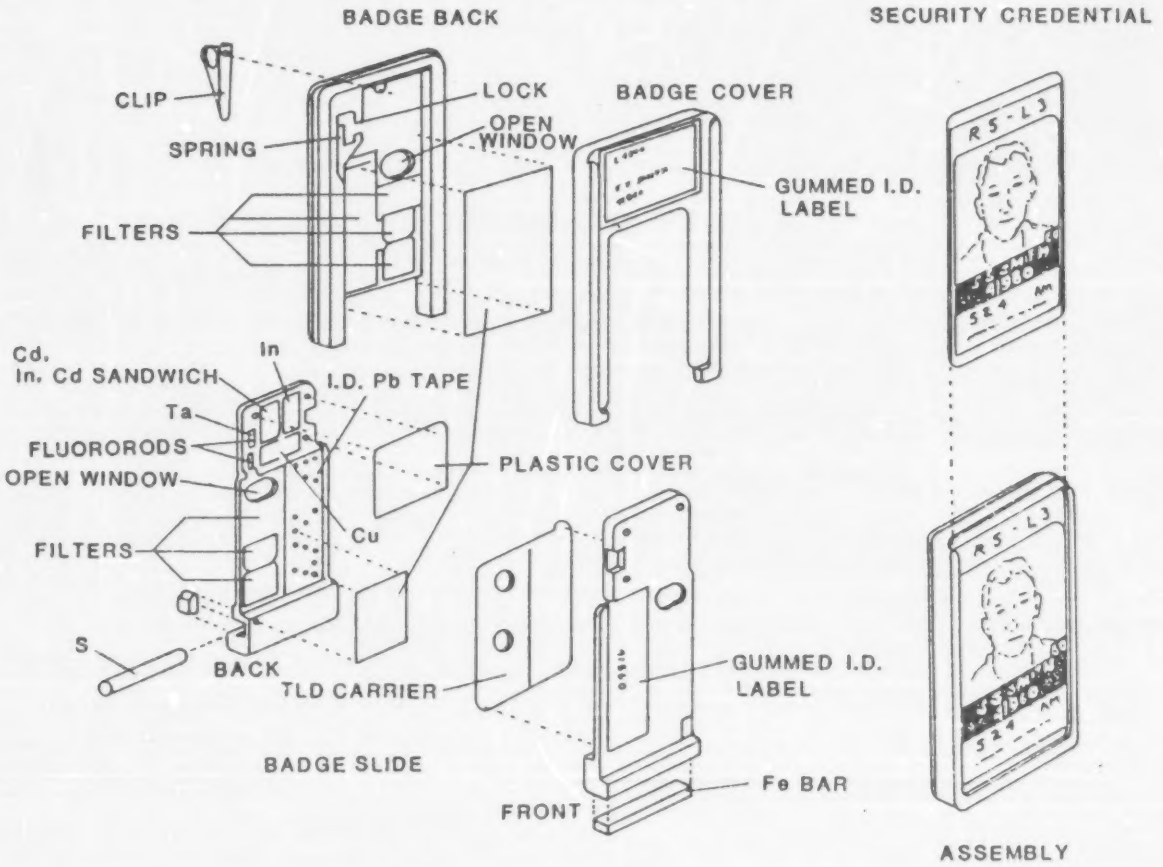


Figure 17-3-4. Typical dosimeter badge that contains fluororods. (This badge, utilizing TLD ir teflon for routine dosimetry, contains two fluororods, one bare and one with flattening filter, for high-level photon dosimetry. It also contains a number of activation foils.)

The typical fluororod measured 6 mm (0.25") in length and about 1 mm (1/25") in diameter. It has a range of from a few rads to about 10^5 rads for gamma radiation.

RPL glasses have certain desirable properties. They are linear; that is, each increment of dose provides an increment of RPL over a very broad range, usually 4 to 5 orders of magnitude. As is the case with any glass, RPL dosimeters are highly resistant to their environment. Thus, they are not adversely affected by temperature or humidity, although very high temperatures (above 100°C for several hours) may cause some loss or fading of the RPL. Temperatures of a few hundred degrees Celsius can be used to anneal the glass and to permit its reuse. In practice, however, the doses simply are cumulated and the glass seldom, if ever, is annealed. This is possible because of the excellent linearity of response and reproducibility (precision) of RPL glass dosimeters.

RPL glasses also are used to measure high-level fields around large ^{60}Co or ^{137}Cs sources and to map gamma fields from high-intensity sources, including reactors, fuel elements, and accelerators. Since these glasses may contain lithium (Table 17-3-1) with a 945-barn cross section for thermal neutrons, they will respond to neutrons as well as to photons. Thus, they cannot be used in mixed neutron-photon fields unless appropriate corrections are made. Correction can be made by the addition of boron. Either way, RPL glasses, if calibrated, also can be used as thermal neutron dosimeters. Even glasses free of lithium will have a response to neutrons because these contain silver, phosphorous, and other elements that possess significant activation cross sections. Hence, use of RPL glass dosimeters is limited to neutron-free photon fields unless appropriate corrections are made.

Readout of RPL glasses is basically simple. An ultraviolet lamp generates ultraviolet light, which is purified further when it is passed through an ultraviolet filter (Figure 17-3-5). The purified ultraviolet light then is allowed to irradiate the RPL glass, which emits red-orange light with a spectrum that peaks at about 610 nanometers. This light, in turn, is passed through a red-orange filter,

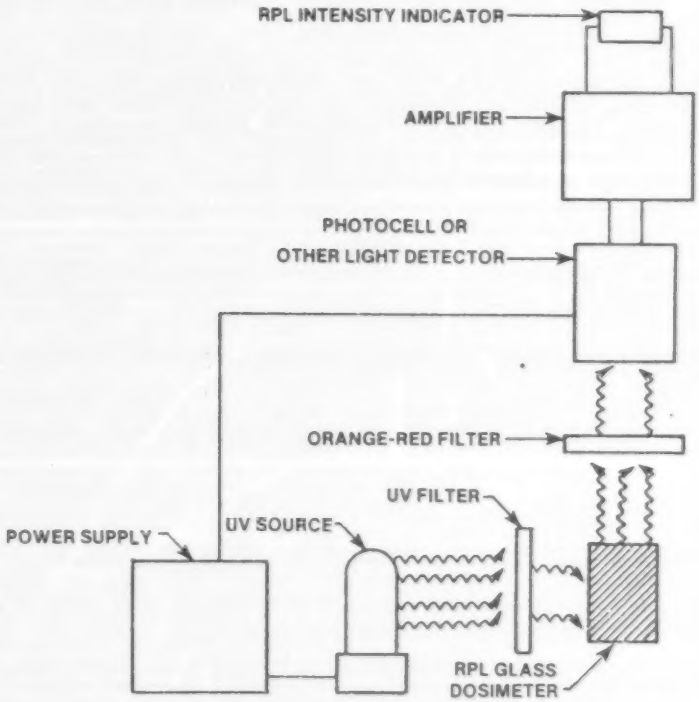


Figure 17-3-5. Schematic drawing of a reader for RPL glass.

which allows only wavelengths of light greater than about 590 nm to pass through. In this manner, scattered ultraviolet or visible light is removed, decreasing interferences and allowing more precise, accurate, and sensitive measurements to be made. The orange-red light is detected by a photocell or photomultiplier, amplified, and read out with

the aid of a meter. It is important that the fluoroglass be clean during the reading, for dust and other material will act to absorb the light and, thus, to reduce the amount of RPL.

TRACK-ETCH DOSIMETERS

In recent years, a new system of dosimetry, known as "track-etching" or "track-etch dosimetry," has become popular. Track-etch dosimetry is useful for neutrons and is based upon the damage done to various materials by charged particles such as protons or fission fragments produced by neutron irradiation. A track-etch dosimeter is really quite simple, as is the readout technique.

For convenience, track-etch detectors can be divided into two groups: (1) those used with fission foils, and (2) those in which tracks are produced as a result of direct interactions with fast neutrons. In the first type, a thin sheet of fissionable material (e.g., ^{238}U) is placed next to a suitable material. The fission foil must be in contact with the material since fission will occur when it is exposed to neutrons and since the heavy, positively-charged fission fragments thus produced do not travel very far. Those fragments produced at, or very near, the surface of the fission foil in contact with the track-etch material may damage the surface of the track-etch material. This damage can be amplified if the material is soaked in an alkali such as sodium hydroxide, which will etch, or eat away, the material around the point of damage, producing tiny pits. The pits (Figure 17-3-6) can be seen easily with the aid of a microscope and can be counted. The number of pits per unit area is proportional to the neutron dose.

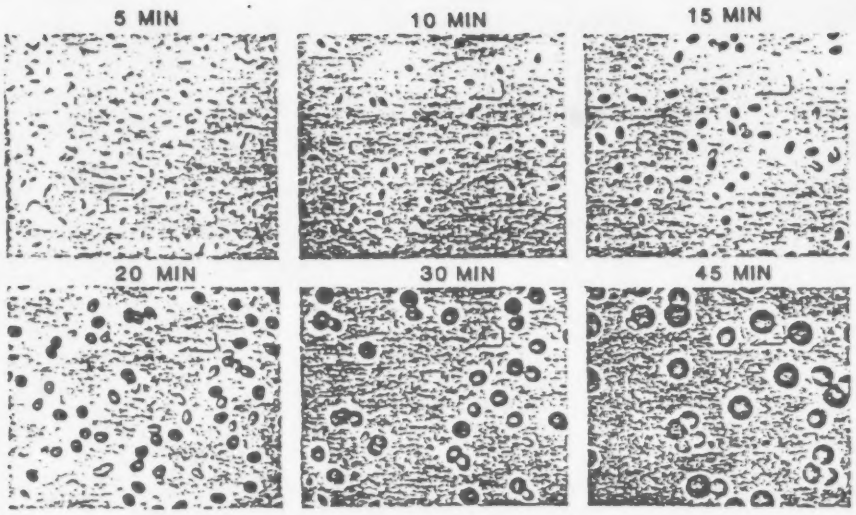


Figure 17-3-6A. Fission-fragment tracks in a phosphate glass as a function of etching time in 28% NaOH at 60°C, viewed through optical microscope.

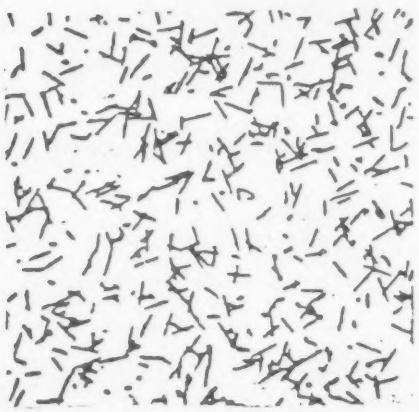


Figure 17-3-6B. Fission-fragment tracks in polycarbonate plastic.

Many materials – including glass, mica, quartz, and various polycarbonate and cellulose nitrate plastics – are suitable track-etch detectors for use with fission foils. RPL glasses covered with tiny fission foils are excellent track-etch detectors, and a single RPL glass can be made to

detect fast neutrons as well as beta, photon, and thermal neutron radiations. Unfortunately, sensitivity of the RPL glass is different for each of these radiations. Otherwise, RPL glass would be an excellent, or even near perfect, dosimeter. Some facilities in Europe, however, utilize dosimeters that contain several pieces of glass and that are useful for a mixture of radiations.

Several different fission foils can be employed with track-etch materials, but all have significant drawbacks. Some foils, such as ^{239}Pu and ^{235}U, are excellent, but are impractical because of high cost, high radiotoxicity, and the fact that these materials are fissile and may require accountability. Foils such as ^{237}Np, ^{238}U, and ^{232}Th do not exhibit these problems to such a great extent, but are useful only for fast-neutron measurements. Figure 17-3-7 gives the fission cross section for these three materials; the best (^{237}Np) has the largest cross section and also the lowest threshold (about 0.6 MeV). The ^{237}Np foil is the most expensive and radiotoxic of the three, and it emits significant external radiation, which could expose the wearer if it were made into a personnel dosimeter. The ^{237}Np foil (as do the other fission foils) undergoes spontaneous fissions that may produce a high background or large number of tracks. Alpha particles also may produce many tracks from their normal decay, even when there has been no exposure to neutrons. For these and other reasons, track-etch dosimeters that contain fission foils seldom are used.

More common, although still largely in the development stage, is the second group of track-etch materials — those in which tracks are produced as a result of direct interactions with neutrons. These materials are plastics and other hydrogenous materials such as mica. $(H_2KAl_3(SiO_4)_3$

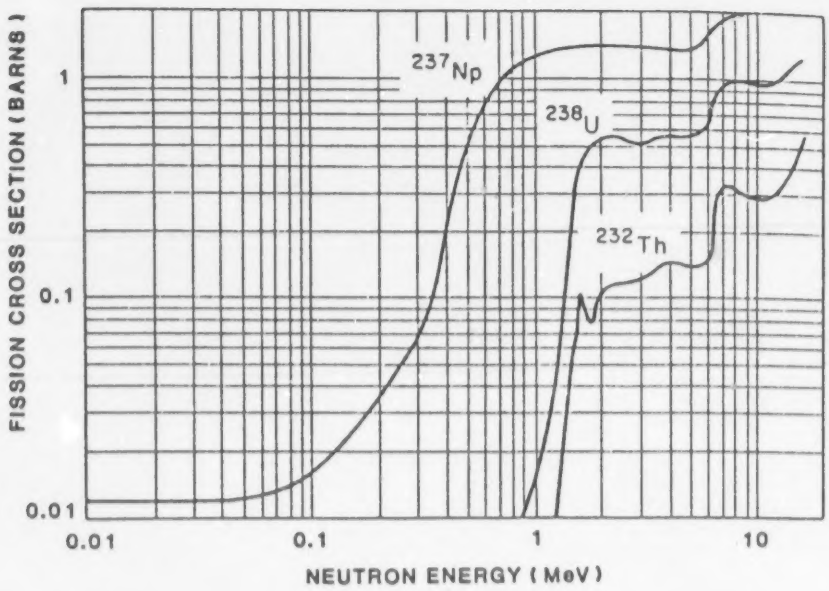


Figure 17-3-7. Fission cross section of different fast-neutron threshold detectors.

and $H_2KMgSi(SO_4)_3$ are the chemical formulas for the two most common hydrogenous micas – muscovite and phlogopite.) In these substances, fast neutrons interact by elastic collisions to knock loose a hydrogen nucleus or to produce a proton recoil. The protons, being positively charged, produce damage centers as they travel through the material. The range of the proton recoil is, of course, quite short; but those produced at, or near, the surface leave damage areas that can be etched.

The advantages of the second class of track-etch detectors are many. They are inexpensive; they use common materials; and they are unaffected by other radiations. Even very high gamma doses do not affect the track-etch dosimetry of neutrons. To count the number of tracks under a microscope

by eye is unnecessary. Automatic track counting can be done simply and easily by spark-counting techniques.

In spark counting, the etched track material (e.g., a wafer of polycarbonate plastic that already has been soaked in alkali) is rinsed, dried, placed on an electrode, and covered with a piece of aluminized Mylar®. The side of the Mylar® that contains the aluminum contacts the etched plastic and also makes contact with ground. When a positive voltage of about 500 V is applied to the aluminum, sparks will occur at the etched pits in the material. These sparks can be counted by an optical device or by an electronic spark counter connected to a scaler. Each pit produces only one spark since the spark results in removal, by evaporation, of the aluminum over the pit, preventing production of a second spark.

Although track-etch dosimetry has certain advantages, including resistance to environmental effects and good linearity, it also has certain drawbacks. These drawbacks include a lack of sensitivity: usually, a track-etch dosimeter has a lower limit of detectability of about 1 rad, or perhaps as low as a few hundred millirads. Two other disadvantages are variable precision, particularly at low doses, and, most importantly, energy dependence. Neutron energy dependence for the hydrogen-containing materials is about the same as that of NTA or other films, in which the production of tracks results from proton recoils. There is a threshold of about 0.5 MeV for the production of a proton recoil. Therefore, track dosimeters such as polycarbonate plastics are insensitive to neutrons in the intermediate and epithermal energy ranges, which greatly limits their usefulness. As was mentioned above in connection with Figure 17-3-7, a similar condition prevails with fission-fragment track-etch dosimeters.

Also, because track-etch dosimeters usually are small, flat sheets, they show pronounced angular dependence (Figure 17-3-8). Directional or angular dependence also is increased in fission-fragment dosimeters worn on the body because of effects of neutron absorption and scattering within the body.

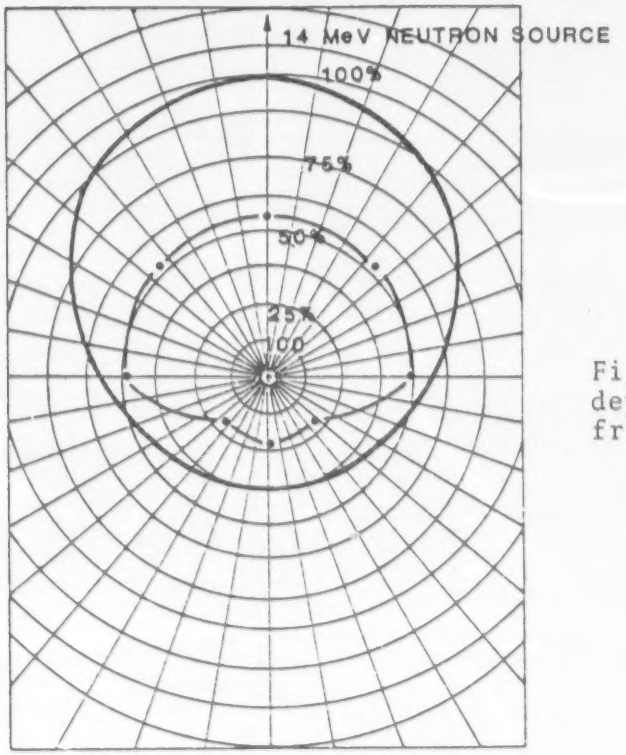


Figure 17-3-8. Angular dependence of a fission-fragment dosimeter.

Despite their limitations, fission-fragment dosimeters are being studied acti ely as potential replacements for NTA film. Their advantages over film include resistance to the environment, which allows a long wearing interval; somewhat easier processing; measurement of neutrons in the presence of photon radiations; and a much broader detection range. Typically, a track-etch dosimeter could provide fast neutron

dosimetry over the dose range of from a few tenths of a rem to several hundred rem. Sensitivity can be increased further if larger areas are counted for tracks or pits. Track-etch dosimeters are thus highly promising for personnel neutron dosimetry.

There are other possible applications for track-etch dosimeters. By use of a track-etch dosimeter such as mica covered with a boron foil, thermal neutrons can be detected from the $^{10}_{5}B(n,\alpha)^{7}_{3}Li$ reaction. In this reaction, alpha particles carry away most of the kinetic energy and penetrate the track-etch material, producing developable track centers. ^{6}Li also can be used, taking advantage of the 945-barn cross section for the (n,α) reaction. The fast neutron response of such dosimeters is not affected by addition of the foil.

HIGH-RANGE AND EMERGENCY DOSIMETERS

Although the likelihood of an accidental high-level exposure to radiation may be small, personnel who work at locations at which this likelihood exists often are issued special high-range, or emergency, dosimeters. If the accidental exposure could be to photons only, a special dosimeter might not be necessary, for TLD's have a range well beyond that of medical concern. Or, a special "backup" TLD or RPL glass rods could be used.

Frequently, however, accidental exposure can involve neutrons. This would be the case in a criticality accident in which a significant fraction of the dose could be from neutrons. After an accidental high-level exposure, the dose

received by the person must be determined as quickly as possible. To accomplish this, personnel accident dosimeters have been devised. These units make use of TLD or RPL glass for the photon exposure, and of a variety of activation foils to measure the neutron exposure. Since they are designed to be worn or carried by people, they must be small and light-weight. The basic design principles of personnel dosimeters for accidental high-range exposures are as follows:

- They must respond to the types of radiation to which the individual might be exposed.
- The package must be small and light in weight so it can be carried easily and conveniently by personnel.
- They must be rugged, and capable of operation, even after being dropped or receiving shocks from the types of anticipated accidents.
- Hazardous materials such as fission foils must not be included.
- They must be capable of easy and rapid processing.
- They should be inexpensive in order that they can be issued in mass quantities without regard to cost.

A typical such dosimeter is illustrated in Figure 17-3-9. This dosimeter contains RPL glass for high-range photon dosimetry, and neutron activation materials of several types. More than one such material is required because no one neutron activation material can be used to measure the entire spectrum of neutrons that would be encountered in a criticality or reactor accident.

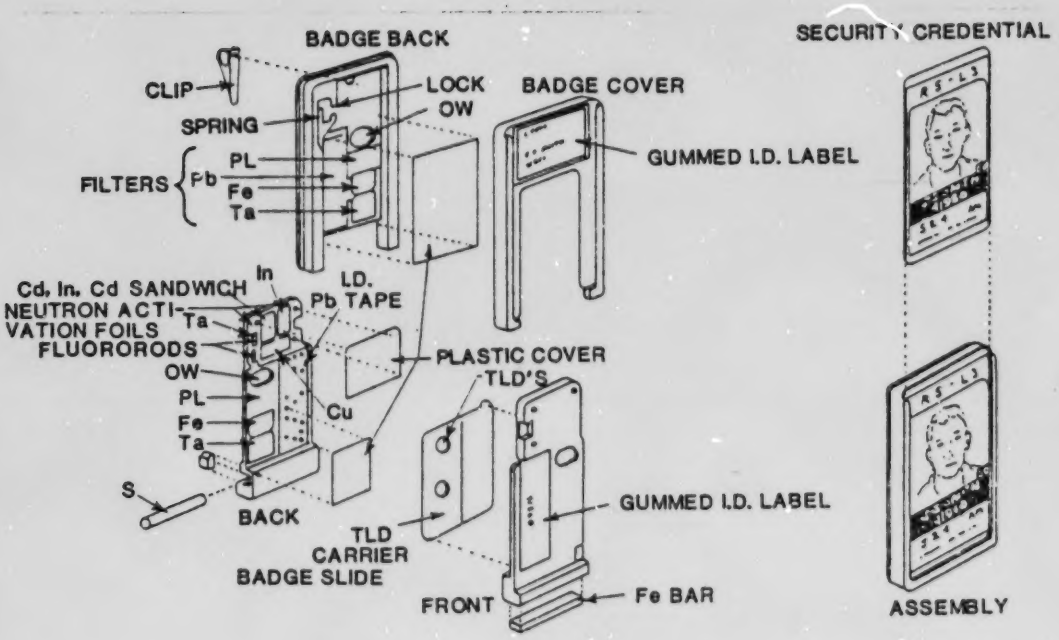


Figure 17-3-9. Typical personnel dosimeter for accidental high-range exposure.

For convenience, neutrons can be divided into several broad regions by energy. These classifications are given in Table 17-3-2, along with their quality factors and the approximate fluence to provide a dose of 1 rem. Different activation materials are employed to detect neutrons of different energy. The choice of material depends primarily upon the cross section or probability of interaction to produce the activated nucleus. Many materials are activated readily by thermal neutrons. Gold, cadmium, and indium have high cross sections and commonly are used since the activation products are nuclides that have reasonable half-lives.

TABLE 17-3-2. CLASSIFICATION OF NEUTRONS

Category	Energy Range	QF	Fluence for dose of 1 rem	Remarks
Thermal	0.025 eV	3	9.6×10^8	In temperature equilibrium
Intermediate	0.5 eV to 10 keV	5	$\sim 4 \times 10^8$	
Fast	10 keV to 10 MeV	10	$\sim 3 \times 10^7$	
Relativistic	> 10 MeV	7.5	$\sim 1.5 \times 10^7$	

Gold, the most commonly used material, undergoes the following reaction to produce a radioactive isotope of gold with a 2.7-day half-life:

$$^{197}_{79}\text{Au} + ^{1}_{0}\text{n} \rightarrow ^{198}_{79}\text{Au} \qquad \text{Equation 17-3-1 (Equation = Eq.)}$$

This is a convenient half-life for counting; ^{198}Au emits a beta particle that can be counted, and the 410-keV gamma ray can be counted.

The relationship of activity and fluence is given by Equation 17-3-2 for a burst, or an exposure to neutrons over a very short duration.

$$A\lambda = \frac{N\sigma\Psi e^{-\lambda t}}{f} = \text{disintegration rate} \qquad \text{Eq. 17-3-2}$$

where:

A = number of atoms activated.

λ = decay constant of the nuclide produced.

N = number of atoms of the original material.

σ = cross section, in cm^2.

Ψ = thermal neutron fluence.

t = time after exposure that the material was counted.

f = efficiency of the counting system.

The following example demonstrates how the dose can be obtained.

EXAMPLE

Given: A dosimeter containing 0.5 g of gold is exposed to an unknown fluence of thermal neutrons. The counting rate on a system with 40% efficiency is 120,000 counts per minute one day after the exposure. The thermal neutron cross section for gold is 98 barns, and the half-life of ^{198}Au is 2.7 days. Given values for Equation 17-3-2:

$$A\lambda = 600{,}000 = 6 \times 10^5 \text{ counts/min}$$

$$\lambda = \frac{0.693}{T_{\frac{1}{2}}}$$

$$= \frac{0.693}{2.7 \text{ day} \times 1440 \text{ min/day}}$$

$$= 1.8 \times 10^{-4} \text{ min}^{-1}$$

$$N = \frac{0.5 \text{ g} \times 6 \times 10^{23} \text{ atoms/g-at wt}}{240 \text{ g/g-at wt}}$$

$$= 1.3 \times 10^{21} \text{ atoms}$$

$$\sigma = 98 \text{ barns} = 9.8 \times 10^{-23} \text{ cm}^2/\text{atom}$$

$$f = 40\% = 0.4 \text{ counts/disintegration}$$

$$t = 1 \text{ day} = 1440 \text{ minutes}$$

Find: Dose received.

Solution: Use Equation 17-3-2.

$$A\lambda = \frac{N\sigma\Psi e^{-\lambda t}}{f}$$

$$6 \times 10^{5} \frac{\text{counts}}{\text{min}} = \frac{1.3 \times 10^{21} \text{ atoms} \times 9.8 \times 10^{-23} \frac{\text{cm}^2}{\text{atom}} \times \Psi \times e^{-1.8 \times 10^{-4} \text{ min}^{-1} \times 1440 \text{ min}}}{0.4 \text{ counts/disintegration}}$$

Solve for Ψ:

$$\Psi = \frac{6 \times 10^{5} \times 0.4}{1.3 \times 10^{21} \times 9.8 \times 10^{-23} \times e^{-0.26}}$$

$$\Psi = 2.4 \times 10^{6} \text{ n/cm}^2$$

According to Table 17-3-2, this value corresponds to a dose of only 2.5 mrem.

Not all foils are as sensitive. Indium and copper frequently are used to obtain the intermediate-energy neutron fluence. These have much smaller cross sections, and since fewer neutrons per cm^2 are required to produce a dose of 1 rem, they are less sensitive on this basis, as well. Fewer neutrons per cm^2 per rem results in fewer disintegrations per minute. For high-energy neutrons, sulfur is employed to determine the fluence. The reaction ${}^{32}_{16}S(n,p){}^{32}_{15}P$ has a threshold at 2.9 MeV, which means that the cross section for the reaction is very small or nearly zero for neutrons having energies below 2.9 MeV. Sulfur, thus, is called a "threshold detector" because no activation results until the neutron energy is above a certain value. A list of common threshold detectors useful for accident dosimetry is given in Table 17-3-3.

TABLE 17-3-3. THRESHOLD DETECTORS USEFUL FOR ACCIDENT DOSIMETERS.

Reaction	Threshold	Cross Section	Half-life of Product
^{237}Np (fission)	0.9 MeV	1.5 barns	Many
^{238}U (fission)	1.9 MeV	0.6 barns	Many
$^{115}In(n,\gamma)^{115m}In$	1.5 MeV	0.35 barns	4.5 hours
$^{31}P(n,p)\ ^{31}Si$	2.5 MeV	0.08 barns	2.6 hours
$^{32}S(n,p)\ ^{32}P$	2.9 MeV	0.3 barns	14.3 days
$^{24}Mg(n,p)\ ^{24}Na$	6.3 MeV	0.05 barns	15 hours
$^{58}Ni(n,p)\ ^{58}Co$	5.0 MeV	1.23 barns	71 days
$^{27}Al(n,\alpha)\ ^{24}Na$	8.6 MeV	0.11 barns	15 hours
$^{28}Si(n,p)\ ^{28}Al$	6.1 MeV	0.19 barns	2.3 min

In addition, some elements show very high cross sections for neutrons in a very narrow energy band. For example, ^{115}In, which has a 145-barn cross section for thermal neutron activation, has a cross section of 28,000 barns for activation by neutrons with energies of about 1.45 eV, the reaction being $^{115}In(n,\gamma)^{116}In$. Gold-197, with a thermal neutron cross section of 98 barns, has a cross section of 29,500 barns for activation by 4.9-eV neutrons. And, sodium-23, with a cross section of less than a tenth of a barn for neutrons above 5 keV and below 1 keV, has a cross section several times greater for neutrons with energies of 3 keV. These peaks in cross section are known as "resonances" or "resonance peaks" and can be used to good advantage in neutron dosimetry because of the greatly increased sensitivity they provide.

Activation analysis dosimetry is a highly complicated and difficult task. The geometry of the detector, changes

in cross section with energy, and presence of resonances are but a few of the difficulties associated with neutron activation analysis dosimeters. Some of these problems can be overcome by use of a person as a dosimeter. People contain many elements useful for neutron activation analysis dosimetry. Sodium, for example, is a normal constituent of blood, and is activated by neutrons according to the reaction $^{23}Na(n,\gamma)^{24}Na$. This reaction has a 0.53-barn cross section. Thus, after an accidental exposure to a high dose of neutrons, a sample of blood can be drawn and counted for ^{24}Na. The results will indicate the average thermal neutron exposure of the body. They will not, however, indicate whether or not only a portion of the body was exposed to a beam that activated the blood sodium as the blood flowed through that portion being exposed to neutrons. Nor will they tell much about the dose, since only thermal neutrons are detected by blood sodium activation.

Some indication of the dose to various parts of the body can be obtained by examination of the hair. Hair and the nails contain relatively large amounts of sulfur, which is activated to ^{32}P by fast neutrons. This reaction has a threshold of 2.9 MeV. If small samples of hair are collected from over the entire body, a map of exposure can be made. Thus, the $^{32}S(n,p)^{32}P$ reaction in hair can determine such information as the direction from which the neutrons came and the parts of the body that received the greatest exposure.

In addition to sodium, iron, and sulfur in the body, other materials such as gold or silver in the teeth or in jewelry worn by the person exposed can be used in an emergency to aid in the estimation of the neutron dose and to determine the direction and size of the beam. Even coins in the pocket, belt buckles, and metallic buttons have been used.

Indeed, the body and its clothing may prove to be the best neutron dosimeter available following an accident.

Many elements are useful for activation dosimetry; Table 17-3-4 lists some of the more important ones. When used as dosimeters, these elements are formed into thin sheets, or foils. The sheet form minimizes attenuation of the neutron beam as well as self-absorption in the material when it is counted. Detection can be by β counting, or by spectroscopy.

TABLE 17-3-4. IMPORTANT ACTIVATION REACTIONS FOR NEUTRON DOSIMETRY*

Nuclide	Percent Abundance	Reaction	Thermal Neutron Cross Section	Half-life of Product	γ Energies
^{27}Al	100	$^{27}Al(n,\gamma)^{28}Al$	0.21 barns	2.3 days	1.78 MeV
^{63}Cu	69.1	$^{63}Cu(n,\gamma)^{64}Cu$	4.5 barns	12.9 hours	1.34 MeV
^{65}Cu	31.9	$^{65}Cu(n,\gamma)^{66}Cu$	1.8 barns	5.1 minutes	1.04,0.83 MeV
^{164}Dy	28.2	$^{164}Dy(n,\gamma)^{165}Dy$	510.0 barns	2.4 hours	Many
^{164}Dy	28.2	$^{164}Dy(n,\gamma)^{165m}Dy$	2100.0 barns	1.3 minutes	95,175, 270,360 keV
^{113}In	4.3	$^{113}In(n,\gamma)^{114m}In$	56.0 barns	50 days	190,552, 722 keV
^{115}In	95.7	$^{115}In(n,\gamma)^{116}In$	155.0 barns	54 minutes	Many
^{54}Fe	5.8	$^{54}Fe(n,\gamma)^{55}Fe$	2.8 barns	2.7 years	220 keV
^{197}Au	100.0	$^{197}Au(n,\gamma)^{198}Au$	98.0 barns	65 hours	412 keV
^{26}Mg	11.0	$^{26}Mg(n,\gamma)^{27}Mg$	0.27 barns	9.5 minutes	0.834, 1.015 MeV
^{107}Ag	51.8	$^{107}Ag(n,\gamma)^{108}Ag$	45.0 barns	2.4 minutes	0.62, 0.43 MeV
^{109}Ag	48.2	$^{109}Ag(n,\gamma)^{110m}Ag$	3.2 barns	260.0 days	0.656, 0.76, 0.935 MeV

*Threshold reactions given in Table 17-3-3.

SPECIAL-PURPOSE DOSIMETRY AND DOSIMETERS

There are some situations, both routine and following an accident, which lend themselves to special measurements. In routine operations, it is frequently desirable to measure dose to the hands since exposure to this portion of the body is likely to be greater. A variety of rings and wrist badges containing film or TLD are utilized for this purpose. One such ring is illustrated in Figure 17-3-10. There are many other styles and kinds of rings. Most of them are constructed of low-Z material, and are small and comfortable. The ring usually is worn on the finger expected to receive the highest exposure, with the portion containing the dosimeter pointing toward the palm. Although TLD is the most commonly used dosimeter for finger rings, some rings utilizing film or RPL glass rods also are worn.

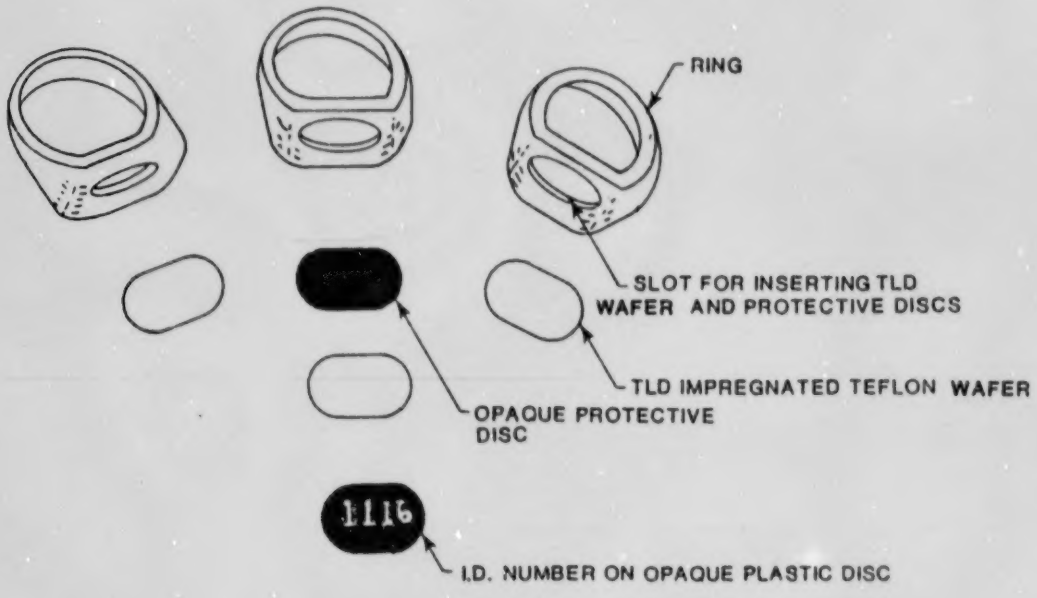


Figure 17-3-10. Finger rings containing impregnated teflon for TLD dosimetry.

Finger ring dosimeters are not highly accurate, and their results should be used only as estimates of exposure to the hands. In addition to other sources of error, such as energy dependence, error may be introduced by the position of the dosimeter on the hand; by shielding from the fingers; and, most importantly, by the fact that dose to hands is seldom uniform.

Occasionally, wrist badges are worn to monitor exposure to the hands and forearms. These units are simply ordinary personnel dosimeters with a wrist strap. Their advantage is that they are larger than a finger ring and contain several TLD chips or a standard-size piece of film. Filters can be included to correct for energy dependence. Wrist badges do not interfere with motion of the hands and fingers. They usually are more comfortable if gloves must be worn. Unfortunately, wrist badges do not give an accurate estimate of dose to the hands. The hand dose normally is about four times greater than that received by the wrist; therefore, the wrist badge reading usually is multiplied by four to obtain a more realistic estimate of hand dose.

Other special-purpose dosimeters include small versions of the pocket ionization chamber equipped with special chips that can be attached to the fingers to obtain the gamma exposure. Sometimes these units are clipped to the side pieces of eyeglasses to gain information on the dose to the lens of the eye, which is a particularly radiosensitive tissue.

Certain photographic films and TLDs can be used to monitor exposure to radioactive gases such as radon and tritium. These dosimeters are designed to allow free interchange of air through their wrappings. The air containing the radioactive gas will expose the dosimeter upon contact. These dosimeters are responsive to external photon, and sometimes to beta, radiations.

EXERCISES

1. Calculate the fluence of fast neutrons to which a sulfur pellet was exposed if the counting rate was 79,212 counts in 10 minutes and if the following were noted:

   Counter efficiency: 31%.

   Time after irradiation: 2 days.

   Use Equation 17-3-2 and the data in Table 17-3-3.
2. From the answer in Exercise 1 and from Table 17-3-2, determine the dose to the dosimeter.
3. Prepare a table that lists all the important characteristics of personnel dosimeters (Hint: Accuracy, precision, cost, energy response, etc.), and compare photographic films, TLD, and RPL glass. Which is the best? Why?
4. The commercial low-Z glass in Figure 17-3-2 was exposed to 100-keV photons from an X-ray machine. If the exposure was 1 R, would the reading be higher or lower than a 1-R exposure of the same glass to ^{60}Co (1.25 MeV photons)?
5. From Table 17-3-4, select a material that could be utilized to detect a neutron dose 2 weeks after the exposure. Why were the particular material and reaction selected?

MATERIALS

5 pocket dosimeters, direct- or indirect-reading type.
Gamma calibration source of known activity.
Dosimeter calibration holder.
Access to calibration range.
Dosimeter charger-reader (if indirect type dosimeters are used).
Dosimeter charger (if direct type dosimeters are used).

PROCEDURES

INTRODUCTION

The personnel pocket dosimeter is a valuable aid to the radiation protection officer in assessing the accumulated radiation exposure of personnel on a daily basis. Exposure history can be utilized in the altering of work assignments where necessary to reduce and minimize personnel exposure. Since the dosimeter must be a reliable source of data, the need for periodic calibration and testing becomes an integral part of the overall radiation protection program.

DOSIMETER CALIBRATION

1. Calculate the present activity of the calibration source.
2. Determine distances from the source that will yield exposure rates of 15, 30, 50, 100, and 105 mR/hr. A dosimeter is required for each of these five rates.

3. Zero and record the serial number of each dosimeter prior to calibration. (If, in the case of direct-reading dosimeters, a true zero value cannot be achieved, record the starting value of the dosimeter.)
4. Mount a dosimeter in a calibration holder and place downrange from the source at a suitable distance to achieve one of the above exposure rates.
5. Expose the source and note the time. Leave the dosimeter exposed to the source for 1 hour. A safety officer should be on hand to insure against unwarranted personnel exposure.
6. When exposure period is complete, secure the source, and retrieve and read the dosimeter. Record readings in Data Table 17-3-1.
7. Repeat Steps 4 through 6 to expose all dosimeters at all designated exposure rates.
8. Evaluate data as follows:
   a. Determine the average %-error of each dosimeter by determining the difference between the sum of expected readings (15 + 30 + 50 + 100 + 150) and the sum of observed net readings. Divide this value by the sum of the expected readings, and multiply by 100. Record this end value for each dosimeter in Data Table 17-3-1.
   b. Rejection criteria is ±10%. Any dosimeter having an average %-error greater than 10% should be discarded or sent for repair.

LEAK TEST DETERMINATION

1. Zero all dosimeters, and record serial numbers in Data Table 17-3-1.
2. Leave dosimeters in an area removed from high radiation levels for a period of 24 hours.
3. After the 24-hour period, read each dosimeter to determine the extent of "charge-leakage."
4. Determine and record the %-leakage against full scale reading of the dosimeter. Rejection criteria is 10% of full-scale readings.

DATA TABLE

DATA TABLE 17-3-1
DOSIMETER CALIBRATION AND LEAK TEST

Date: _____

Source: _____ Name: _____

Dosimeter No.	Dosimeter Reading					Average Percent Error	24 hr Leak Test	Passed	Failed
	15 mR	30 mR	50 mR	100 mR	150 mR				

REFERENCE MATERIALS

Becker, Klaus. Solid State Dosimetry. Cleveland, OH: CRC Press, 1973.

Moe, H. J. Radiation Safety Technician Training Course. ANL-7291, Rev. 1, 1972.

Peisch, E. "Developments in Radiophotoluminescence Dosimetry," Topics in Radiation Dosimetry, F.H.A. Hix, Ed. New York: Academic Press, 1972.

Yokota, R. and Nakajima, S. "Improved Fluoroglass Dosimeter as a Personnel Monitoring Dosimeter and Microdosimeter," Health Physics 11:41, 1965.

1. Define the following terms:

   Fission foil.

   Fission fragment.

   RPL.

   Track etching.

   NAD.

   Personnel dosimeter.

   External dose.

Penetrating dose.

Nonpenetrating dose.

2. Describe the theoretical basis of radiophotoluminescence dosimetry.

3. Identify and explain applications for finger and wrist dosimeters.

4. Explain the applications and limitations of typical high-range personnel dosimeters.

5. Identify commonly used fission and activation foils utilized in personnel dosimetry; write the nuclear equations for reactions.

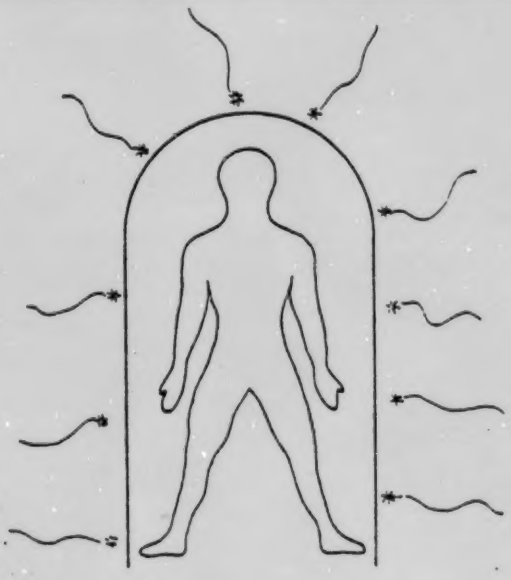
6. Calculate the dose for the following circumstances: A dosimeter that contains 0.6 g of gold is exposed to an unknown fluence of thermal neutrons. The counting rate on a system having an efficiency of 41% is 120,000 cpm one day following the exposure. (The thermal neutron cross section for gold is 98 barns, and the half-life of ^{198}Au is 2.7 days.) Is all the information here given required for the solution of this problem? Why?

7. Describe spark-counting and track-etch detectors.

NUCLEAR TECHNOLOGY

MODULE 17 - 4

INTERNAL DOSE ASSESSMENT



RADIATION PROTECTION II

COURSE 17

CORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 17 RADIATION PROTECTION II

MODULE 17-1	Film Dosimetry
MODULE 17-2	Thermoluminescent Dosimetry (TLD)
MODULE 17-3	Radiation Dose Assessment - Personnel Dosimeters
MODULE 17-4	Internal Dose Assessment
MODULE 17-5	Bioassay Techniques and Recordkeeping for Radiation Protection

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INTRODUCTION

The first three modules in this course have been concerned with external dosimetry – the measurement of dose to persons from radiations originating outside the body. This module will be concerned with the dose received from radioactive elements that undergo their normal radioactive decay within the body.

Each person has literally billions upon billions of radioactive atoms in his or her body. Most of these radioactive atoms are naturally occurring; for example, natural potassium, an element normally found in human bodies and necessary for life itself, consists of a mixture of three isotopes: ^{39}K (93.08%), ^{40}(0.0118%), and ^{41}K (6.91%). The ^{39}K and ^{41}K, which together make up about 99.99% of the natural potassium, are stable. The ^{40}K, however, is radioactive, undergoing beta decay with a half-life of 1.28×10^9 years. This weak radioactive form of potassium is found wherever natural potassium is found – in rocks, soils, plants, animals, and human bodies. Each gram of potassium found in nature, although containing only 0.0118% of the radioactive isotope ^{40}K, undergoes approximately 1,900 beta disintegrations per minute.

The average adult man's body contains about 140 g of potassium, or 140 x 1,900 = 266,000 disintegrations/min – approximately 0.12 μCi of radioactive ^{40}K. This amount produces a dose of 20 mrem per year to the body. The body also contains other radioactive elements found in nature. These elements include ^{14}C, a radioactive isotope of carbon produced in the atmosphere by cosmic-ray interactions on nitrogen and incorporated into all living material, since carbon is one of the fundamental elements of life. Indeed, the average man contains about 16 kg of carbon; an insignificant fraction is the radioactive isotope ^{14}C. On an

activity basis, the amount of ^{14}C is approximately the same as that from ^{40}K – about 0.1 μCi. The dose, however, is only about 0.7 mrem/yr, or about 4% of the dose ^{40}K.

Several other naturally-occurring radioactive elements normally are present in human bodies. On an activity basis, tritium, a radioactive form of hydrogen, is the largest in amount but produces the smallest dose. Tritium in the form of water is present in all cells and tissues, and to a much lesser degree chemically combines with carbon to form radioactive amino acids, fats, and enzymes. Virtually anything in the body that contains hydrogen contains tritium. Also found in the body are trace amounts of other naturally-occurring radioelements – ^{87}Rb, ^{226}Ra, ^{210}Po, and ^{nat}U. These, plus the radioactive gases ^{222}Rn and ^{220}Rn (radon and thoron), inhaled with each breath only add a few millirem to the annual total of about 25 millirem received each year from naturally-occurring radioactivity inside the body.

Why should the doses be so different from approximately equal amounts of ^{40}K and ^{14}C? Why should tritium, present in much greater activity, produce a negligible dose to the body? Why does radium concentrate in the bone and only produce a dose to the bone? These and other questions will be answered in this module, but this introduction should be reread after the entire module is read. Also explained is how specific, artifically-produced radioactive nuclides enter the body and how they produce doses to various tissues and organs. Internal dose equivalents to various body organs are an important aspect of radiation protection. The calculations necessary to determine the doses from radioactive materials that have been taken into the body are known as "internal dose equivalents."

MODULE PREREQUISITES

The student should have completed courses in basic chemistry, algebra, and biology; Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; and Course 11, "Radiation Detection and Measurement."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define the following terms:
   - Maximum Permissible Concentration.
   - Maximum Permissible Body-Burden.
   - Bone seeker.
   - Organ dose.
   - Critical organ.
   - Biological half-life.
   - Standard man.
   - Phagocytosis.
2. Describe the ways in which radioactive material can enter the body.
3. Trace the metabolic pathway of inhaled radioactivity within the body.
4. Explain the importance of the effective half-life, effective absorbed energy, and mass of the organ from the standpoint of internal dose assessment.
5. Calculate dose equivalent to an organ from a quantity of a specific nuclide in the organ.
6. Calculate dose rate to an organ with a quantity of a radioactive material in it at any time.

7. Given biological and radiological half-life, calculate effective half-life.
8. Describe generally how MPCs and MPBBs are determined.

ENTRY OF RADIOACTIVE MATERIALS INTO THE BODY

Radioactive materials can enter the body by three routes: inhalation, ingestion, or through the skin. Entry through the skin is known as "percutaneous absorption" and is the least important from the standpoint of radiation protection. Inhalation, or the breathing in of radioactive active substances, is the most important route of entry.

The breating rate of any person is dictated by the body's need for oxygen. An average man at rest breathes air in at the rate of about 7.5 liters/hour. The same man, engaged in light work or normal activity, will breathe at the rate of 20 liters per minutes. If engaged in very strenuous physical activity, he may breathe more than 100 liters per minute. The quantity of air taken into the body each minute depends upon only two factors: the number of breaths taken and the volume, or size, of the breath. To some extent, these factors depend upon the physical activity that the individual is performing; however, the size, or volume, of the breath also depends upon the size of the individual. An infant, for example, does not have the capacity, or lung size, of an adult; and among adults, women are, on the average, smaller than men. Thus, if there is a certain concentration of radioactivity in the air, the amount taken into the body will depend upon how much air is breathed.

This concept can be demonstrated quite easily by a simple example. Suppose that the air contains 3 pCi/l of radioactive material X. A man at rest, breathing 7.5 liters of air per minute, would inhale 22.5 pCi of X every minute, or, as shown mathematically –

3 pCi/l x 7.5 l/min = 22.5 pCi/min.

The general case is expressed by Equation 17-4-1.

$$A = CV \qquad \text{Equation 17-4-1 (Equation = Eq.)}$$

where:

A = activity taken into the body.

C = concentration of radioactivity in air.

V = volume of air inhaled.

Since there are so many variables that affect breathing rate, a set of standard values based upon average values has been adopted. These are given in Table 17-4-1. In radiation protection work, it is customary to use a value of 20 liters per minute as the breathing rate for an adult under normal working conditions. On this basis, the occupational maximum permissible concentrations (MPCs) or radioactivity concentration guides (RCGs) are calculated. The nonoccupational, or 168 hour/wk values, are based upon an adult breathing rate of 20 liters/min for 16 hours per day and 7.5 liters/min for 8 hours per day.

TABLE 17-4-1
STANDARD BREATHING RATES

Activity	Breathing Rate, Liters Per Minute				
	Adult Man	Adult Woman	10 Year Old	1 Year Old	Newborn
Normal activity or light work	20	19	13	5.8	1.5
Resting	7.5	6	4.8	1.4	0.5

The amount of radioactivity inhaled is not necessarily the amount that gets into the body. Many factors affect how much reaches the lungs, how much is exhaled, how much stays in the lungs, and how much is absorbed into the blood stream. The most important factors are –

- Physical state.
- Particle size and shape.
- Solubility.

The physical state refers to whether the material is in gaseous, solid or liquid form. Generally speaking, 100% of all the radioactive gases inhaled gets into the lungs. Some gases, such as ^{222}Rn, are chemically inert and are exhaled almost completely. Only a very small fraction of the ^{222}Rn inhaled is transferred to the blood. Others, such as tritiated water vapor (HTO), may be completely transferred or absorbed into the blood.

Most airborne radioactivity is particulate or solid in nature. The air we breathe is full of tiny particulates. Even so-called "clean air" contains several thousand particles per liter, and dusty air can contain from several hundred thousand up to a few million particles per liter. Not all of these particles, however, will get into the body; for the respiratory system is equipped with the means by which to remove the larger ones and to prevent others from entering the bloodstream. Before a person can understand how the respiratory tract accomplishes this, some knowledge of the structure or anatomy of the respiratory tract is necessary.

The respiratory tract (Figure 17-4-1) can be divided into three parts, or compartments. The uppermost and outermost of these are nose and throat, or the so-called "naso-pharyngeal compartment," abbreviated N-P. In breathing, air first is taken into the nose (or mouth) and then passes down into the throat or pharynx. The portions of the nose just inside the nostrils are lined with skin that contains hairs, and these hairs act as a filter to remove dust particles having diameters greater than about 10 μm. The remainder of the N-P compartment is lined with cells that contain tiny, microscopic, hairlike structures known as "cilia." The cilia are constantly in motion, moving the mucous secreted upward from the pharynx towards the mouth and downward from the nose toward the pharynx.

From the pharynx, the air passes through the larynx, or voice box, into the tracheo-bronchial compartment, or tree, abbrevicated T-B. The trachea is a tube strengthened by rings of cartilage and measures 1 to 2 cm in diameter. It divides into two bronchi, one going to each lung. The bronchi further divide into smaller tubes, or bronchioles,

which deliver air to various parts of the lung. The tracheobronchial tree is lined with ciliated cells like those that line the N-P compartment, and the mucous layer constantly is moved upward toward the pharynx.

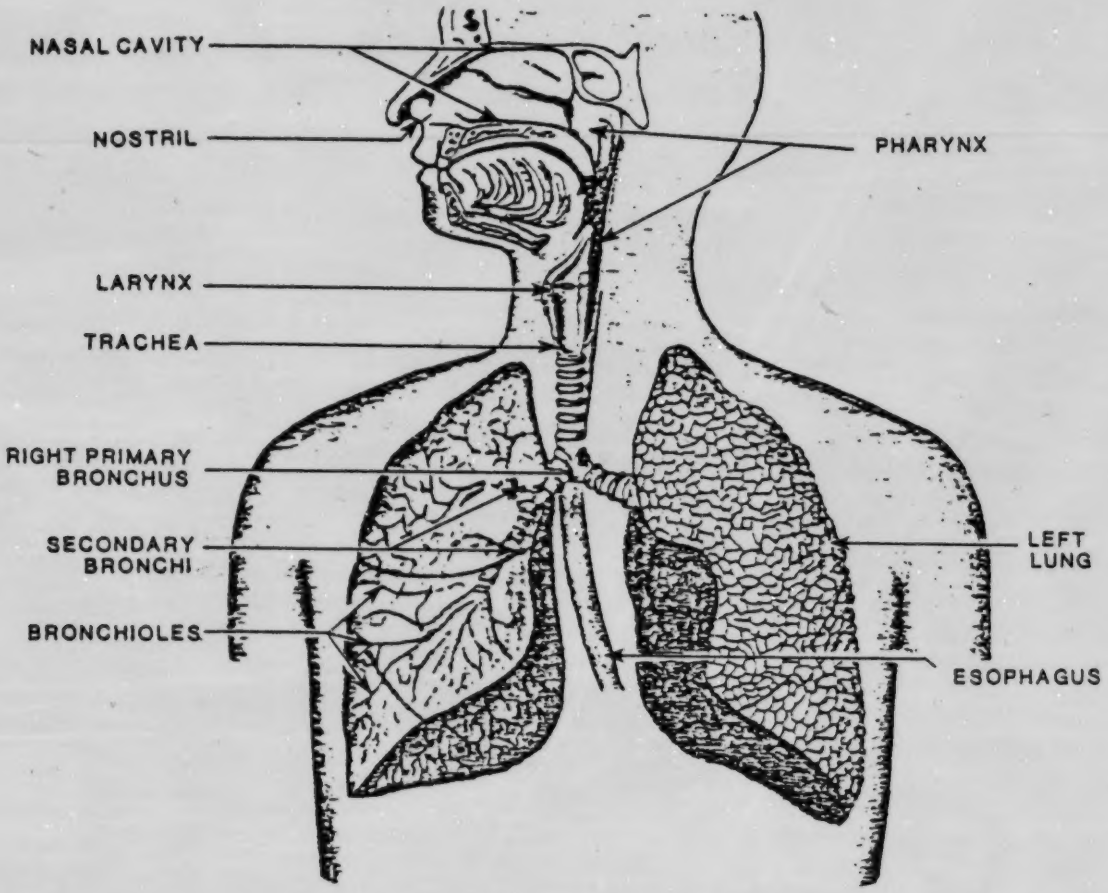


Figure 17-4-1. The respiratory tract.

The third and deepest lung compartment is known as the "pulmonary compartment," abbreviated P. The P compartment consists of the terminal, or smallest, bronchioles that end in a cluster of tiny bag- or sac-like structures

called "alveoli." In the alveoli, respiration (exchange of gases in the blood) occurs. The alveoli, illustrated in the enlarged view in Figure 17-4-2, number more than 700,000,000 in the normal adult and, if spread out, would provide a respiratory exchange area of about 70 m^2. The alveoli do not contain cartilage but are richly supplied with capillaries. Each globelike alveolus measures about 100 μm in diameter.

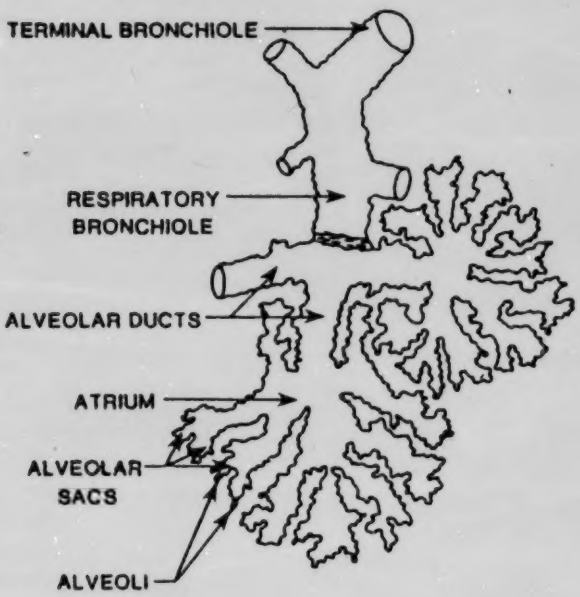


Figure 17-4-2. Enlarged view of anatomical structure of terminal bronchioles and alveoli.

When air is breathed, it first is taken into the nose, where large particles are filtered out by the hairs. Large particles not filtered out in this fashion soon are removed by deposition on the walls of the pharynx. Such particles move with the air stream but have so much inertia, or resistance, to a change in the direction in which they are moving that they cannot move around the sharp bends in the nose and pharynx. Thus, they bump into the walls of the N-P compartment and stick on the mucous layer.

As the air moves farther down the respiratory tract into the T-B compartment, particles with diameters in the region of 1 to 10 μm are deposited or removed. The smaller particles are carried into the P compartment, and those having diameters greater than about 0.5 μm settle into the deep lung spaces by gravity. Smaller particles are deposited on the walls of the alveoli by impaction or actual collison with the structure. Deposition of particles according to size is illustrated in Figure 17-4-3 for each of the three regions and for the respiratory tract as a whole. Nearly 100% of the particles having aerodynamic diameters greater than 10 μm are removed from the air. Particle removal (or deposition in the respiratory tract) decreases as the particles become smaller, and for particles having diameters of about 0.1 μm, deposition is least. Only about 25% of these particles is removed from the air and deposited in

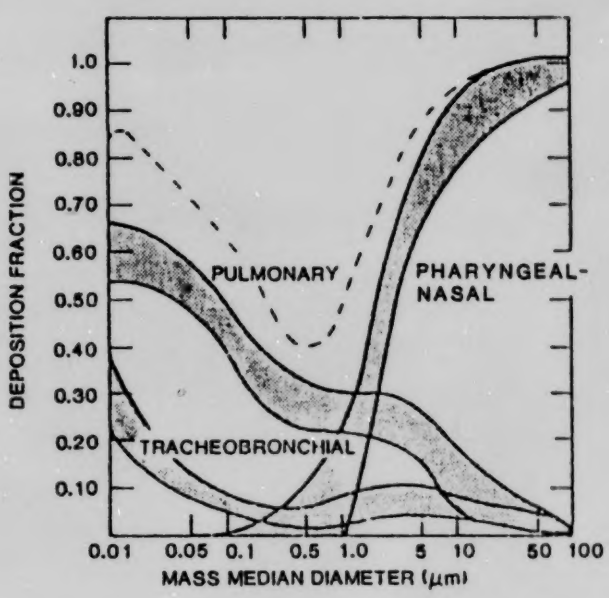


Figure 17-4-3. Deposition of dust in various compartments of the respiratory tract. (Shaded bands represent variability of deposition. The dashed line indicates approximate total in the entire respiratory tract.)

the respiratory tract. A greater percentage of smaller particles is removed, however, and deposits almost entirely in the P compartment, or deep lung spaces.

Retention of particles within the respiratory tract depends upon where they were deposited and upon their chemical and physical properties. Particles deposited in the N-P and T-B compartments are removed by ciliary action in the throat, where they are swallowed. Movement by ciliary action is very swift, ranging from about 10 cm/hr in the bronchioles to about 200 cm/hr in the trachea. The cough and throat-clearing mechanisms aid this clearance. Once swallowed, the radioactivity becomes ingested, and is absorbed as discussed shortly under the heading of "absorption."

Smaller particles that have been deposited in the P compartment may be cleared by one of two mechanisms: absorption into the blood stream or phagocytosis. It is here that solubility of the radioactivity becomes important. Soluble particles dissolve in the lung fluids and move into the blood in the alveolar capillaries. Most inhaled soluble radioactive materials enter the bloodstream by this route and, once in the blood, can be carried to other parts of the body.

Phagocytosis is a removal process by which large cells known as "phagocytes" surround the dust and either remove it via the ciliary pathway or move it into the lymphatic system. Once in the lympathic system, the phagocytes containing the radioactive particles may be trapped in the lymph nodes that run along the sternum (breastbone) and trachea, and the radioactive material either may be held or retained there for years or slowly absorbed into the blood. Phagocytosis, of course, is the main removal mechanism for insoluble particles deposited in the deep lung spaces or P compartment.

ABSORPTION

Absorption of inhaled radioactive materials is highly complex. Radioactive materials can be absorbed into the blood from all three compartments as well as from the gastrointestinal tract. The many possible pathways of an inhaled radioactive particle are indicated in Figure 17-4-4.

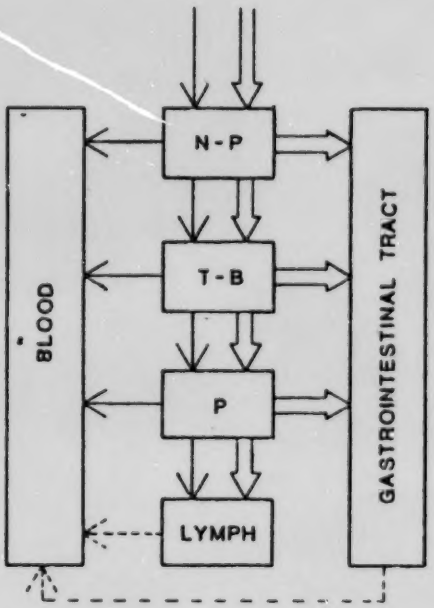


Figure 17-4-4. Schematic representation of lung clearance. (Open arrows refer to particulates normally insoluble in lung fluid; solid arrows to soluble material; and dashed lines to insoluble particulates that have been solubilized elsewhere.)

Ingestion by mouth is the second major way in which radioactive materials can be taken into the body. Depending upon their chemical nature and solubility, these materials commonly are absorbed into the bloodstream through

the small intestine. Insoluble materials may pass right on through the gastrointestinal tract, in which case they will be excreted in the feces. During its transit through the gastrointestinal tract, the radioactive substance is decaying and exposing the gut and other portions of the body to a radiation dose. Ingestion by mouth can be controlled by the restriction of eating, gum chewing, and smoking in areas in which loose radioactivity may exist.

Material cleared from the lungs via ciliary action is swallowed into the stomach. If soluble or made soluble by the digestive process, it will be absorbed into the bloodstream through the small intestine. If insoluble, it will pass out of the body through the feces.

The final way in which radioactive materials can get into the body is through the skin. Some gaseous materials, notably tritium, HTO vapor, and radioiodine, can be absorbed into the blood through the intact (unbroken) human skin. Absorption through intact skin can be prevented, to some extent, by clothing and to a great degree by plastic suits. Plastic suits sometimes are worn, therefore, by workers to protect themselves from certain kinds of airborne radioactivity. An even more significant mechanism of entry through the skin is through wounds such as cuts or through skin that has been traumatized by heat, cold, or chemical action. Percutaneous absorption, for example, is greatly increased through skin that has been in contact with organic solvents such as trichloroethylene or gasoline. These substances defat the underlying skin layers, allowing materials in solution to pass through more readily. Similarly, if the normally acid pH of the skin is changed, absorption may be increased.

Insoluble radioactive material may deposit in a cut or puncture wound. This material, if not washed out, can be solubilized slowly and absorbed into the body over a period of weeks or even years. If the particles are small enough, they may be phaogcytized and transported to nearby lymph nodes.

METABOLISM OF RADIOACTIVE MATERIALS

The term "metabolism" refers to the fate or use of radioactive materials within the body. Radioactive elements chemically behave no differently from nonradioactive ones. Thus, if radioactive calcium is taken into the body, it will behave no differently from stable calcium; for both are identical chemically. Both the nonradioactive and radioactive isotopes of calcium may be incorporated into bone, for that is where most of the calcium in the body is. Once in the bone, calcium atoms will not remain there indefinitely. Every element in the body is periodically turned over or exchanged with new atoms of the same element; the old atoms are excreted from the body, generally in the urine, but they also could be lost through the feces, sweat, or saliva.

An examination of calcium in the body can lead to a better understanding of how a stable element, vital to life, is metabolized. An average adult weighing 70 kg (154 lb) ingests about 1 g of calcium in food and drink every day. Since most compounds of calcium are soluble, there is considerable absorption by the gastrointestinal tract. About 60% of the ingested calcium is absorbed into the bloodstream, and 90% of the calcium in the blood ultimately is incorporated into the bone, either as new growth or to replace

calcium atoms already present. The replaced calcium atoms get into the blood, but are removed by the kidneys and excreted in the urine along with the small fraction of calcium atoms that were absorbed into the blood but not taken up by the bone. The kinetics of the process are highly complicated, and can be fully described only with extensive mathematics, although the process can be examined in a more general and simple way. The following simpler description is adequate for most radiation protection activities.

If the calcium taken into the body over a period of one day for a lengthy period were followed, the calcium could exist in one of several places – or "compartments," as they are called – within the body. From the previous paragraph, these compartments can be identified as –

- Gastrointestinal tract.
- Blood.
- Bone.
- Kidney.
- Urine.

An individual calcium atom would spend a different amount of time in each compartment: a day or two in the G-I tract, perhaps a few hours in the blood, a few years in the bone, a few minutes in the kidney, and perhaps minutes to hours in the urine. A mathematical expression can be utilized to describe movement of the calcium to and from each compartment, and an expression that is a combination of all these equations can be employed to describe or to calculate how fast the body is excreting the material it has taken in.

Ordinarily, a half-life can be used to describe the rate of removal of an element either from the body as a whole or from a specific organ or tissue. Again, with

calcium as an example, the biological half-life of calcium in the body is about 45 years; a period of 45 years is required for the calcium to be removed from the body. The biological half-life for calcium in the bone is even longer. Once the calcium reaches the bone, its clearance or removal will be with a half-life of 49 years. The total body half-life consists of the half-lives for clearance from all the compartments; therefore, it will always be shorter than some and longer than others.

The biological half-life, simply stated, is the time required for half the element to cycle through the body. Typical values of biological half-life are highly variable, and depend upon the chemical nature of the element and where it is found in the body. Table 17-4-2 lists the biological half-life for several important elements, both in the body as a whole and in the organs in which these elements tend to localize in the body. Notice that biological half-life is designated by the symbol T_B.

In addition to removal by biological processes, radioactive elements undergo radioactive decay while in the body. The element, ^{45}Ca, for example, as a radioactive isotope of calcium has a physical half-life of 164 days; it decays to ^{45}Sc, a stable isotope of scandium. Thus, the half-life of ^{45}CA in the body is not 46 years, since it is undergoing physical decay and is disappearing with a half-life of 164 days. Actual half-life of ^{45}Ca in the body is actually determined by the combined effect of both the physical and the biological half-lives. This combination is known as the "effective half-life" (T_{eff}), which can be calculated for any radionuclide by Equation 17-4-2.

TABLE 17-4-2

BIOLOGICAL HALF-LIFE OF IMPORTANT STABLE ELEMENTS

Element	Organ	T_B	% of Element in Organ
K	Total body	58 days	100
	Muscle	58 days	65
Ca	Total body	45 years	100
	Bone	49 years	90
H	Total body	10 days	100
C	Total body	10 days	100
	Body fat	12 days	50
Na	Total body	11 days	100
P	Total body	257 days	100
	Liver	18 days	7
	Bone	3.2 years	20
Mn	Total body	17 days	100
	Liver	25 days	24
	Kidneys	6.8 days	5
	Pancreas	5.7 days	
I	Total body	138 days	100
	Thyroid	138 days	20

$$T_{eff} = \frac{T_R \times T_B}{T_R + T_B} \qquad \text{Eq. 17-4-2}$$

where:

T_{eff} = effective half-life.

T_R = physical or radioactive half-life.

T_B = biological half-life.

Equation 17-4-2 is applied in the solution of a problem in the following example.

EXAMPLE

Given: ^{45}Ca, whose physical or radioactive life is 164 days and whose biological half-life is 46 years (1.64×10^4 days).

Find: The effective half-life of ^{45}Ca.

Solution:

$$T_{eff} = \frac{T_R \times T_B}{T_R + T_B}$$

$$= \frac{164 \times 1.64 \times 10^4}{164 + 1.64 \times 10^4} \text{ days}$$

$$T_{eff} = \frac{2.69 \times 10^6}{1.66 \times 10^4} = 162 \text{ days}$$

Metabolism of any element, radioactive or stable, depends upon its chemistry. Most of the table of more than 100 known elements are not essential for life. Nonetheless, these are taken into the body with food and drink or by inhalation and, if soluble, are taken up by the body in various ways. If, for example, they are chemically similar to calcium, they will be treated by the body much in the same way as calcium. Strontium, an element chemically like calcium, will be incorporated into the bones by the body in a manner very similar to calcium. Thus, the biological half-life for strontium is about the same as that of calcium, and 95% of the strontium in the body is found in the bones. Bones would contain far more Sr except that only 1 mg per

day is ingested, compared with 1 g of calcium. Also, because of the higher atomic weight of Sr, the body seems to be able to discriminate somewhat between it and Ca.

In addition to strontium, many elements behave like calcium chemically and are deposited in the bones. These elements include the other elements of the 2a Group of the periodic table (Be, Mg, Ba, and Ra) and the actinides. The actinides are the group of heavy elements with $Z > 89$, including U, Pu, Am, and Th. All known actinium isotopes are radioactive; many are alpha emitters having very long half-lives. If taken into the body and deposited in the bones, they will remain there for essentially the lifetime of the individual; for their biological half-lives are quite long. The biological half-life of plutonium in bone is 200 years, and its radiological half-life is 24,360 years. Its effective half-life, therefore, is —

$$T_{eff} = \frac{24,360 \times 200}{24,360 + 200} = 198.4 \text{ years.}$$

A person would have to live to be 198 years of age before half the ^{239}Pu in his bones would be removed. If a baby were to have an intake of ^{239}Pu on the day of its birth, even if it matured and lived to be 100 years of age, more than 70% of the plutonium that was deposited in its bones from the intake on the first day of its life would be in the aged bones on the day of death.

The actinides, radium, and ^{90}Sr are very important from the standpoint of internal dose. Because these elements are localized in the bone, they are known as "bone seekers,"

and they are particularly hazardous because the blood cells are formed in the marrow of the bone. These radionuclides, if incorporated into bone, will irradiate the juvenile blood cells at a time when they are particularly sensitive to radiation.

Certain other elements accumulate in other parts of the body. Tritium, when incorporated into water (H_2O), will be spread throughout the body. Iodine is found mostly in the thyroid gland, a small butterfly-shaped organ in the neck. When inhaled in insoluble forms, most elements will accumulate in the lungs or lymph nodes near the lungs. The organ in which the radioactivity accumulates usually receives the highest dose and is known as the "critical organ."

INTERNAL DOSE CALCULATIONS

The dose equivalent delivered to any organ from a radioactive material depends largely upon these factors:

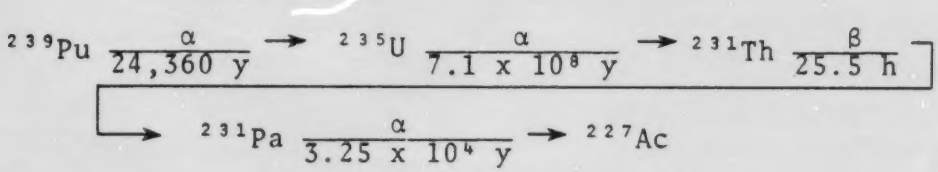
- Size of the organ.
- Amount and distribution of radioactivity in the organ.
- Types and energies of radiations emitted.
- Effective half-life of radioelement.

Effective half-life already has been discussed. Clearly, the longer the effective half-life, the more disintegrations take place in the organ over a given period of time for a given quantity of initial activity. Consider, for example, two isotopes of the same element, one with an effective half-life of 10 years and the other with an effective half-life of 10 minutes. If each initially is 1 μCi in activity,

how many disintegrations will occur in 1 day? For the long (10 year) half-lived isotope, the rate of decay will not change appreciably in one day; and the number of decays will be equal to the decay rate (2.2×10^6 disintegrations per minute) multiplied by the time (1 day or 1440 minutes) for a total of 3.17×10^9 disintegrations. The short (10 minute) effective half-life isotope will be gone in a few hours. It will not produce as many decays in the organ as its brother of long half-life. (The exact number can be determined, with the aid of calculus, as 3.20×10^7 disintegrations.) If the same amount of energy were deposited by each isotope's disintegrations, the dose rate from the shorter-lived one will drop off very, very rapidly, although both begin giving the same dose rate.

Most of the dose equivalent from internally-deposited radionuclide results from alpha and beta particles emitted during decay. Typically, an alpha particle has a range of 500 to 100 μm in tissue and has an energy of several million electron volts associated with it. All of this energy is absorbed very close to the point at which the alpha particle originated. Since dose is simply energy absorbed per unit mass, an alpha particle will produce much energy in a small mass and, hence, a relatively large dose. The same holds true for beta particles, although their range is somewhat larger, and their energy is somewhat lower. Thus, the average amount of energy deposited per unit mass — in other words the dose — is much lower. Lowest of all is the dose from gamma rays, which travel relatively large distances before interacting and giving up energy. For all practical purposes, the gamma-ray contribution to internal dose is negligible because most gamma rays escape from the organ and, perhaps, from the whole body without interacting.

For most radionuclides, the effective absorbed energy deposited in tissue has been calculated. The effective absorbed energy is the sum of all the energy from all the particles, X-rays, and gamma rays absorbed from a particular nuclide in a particular tissue. Also included in the effective absorbed energy value is the quality factor for the various radiations. The effective absorbed energy usually is designated by the symbol ε, or in older literature $\Sigma E(RBE)n$. It includes not only the energy from the decay of the parent nuclide, but also that from any radioactive daughters. Importance of the radioactive daughters can be great, particularly among the bone-seeking radionuclides. Consider ^{239}Pu, for example. The first part of the ^{239}Pu decay chain is as follows:



The five nuclides that start the chain are actinides. Thus, even if the ^{239}Pu atom decays it will produce a new alpha-emitting bone seeker, namely ^{235}U. Because of the alpha particles produced by decay of the actinides, value of the effective absorbed energy, $\Sigma(RBE)n$, can be quite large; for alpha particles have a QF of 10. In addition, the n refers to a special weighing factor of ε, which is applied to all bone seekers except radium. This is a relative damage factor and is applied to the energy associated with the alpha and beta decay. (Radium is deposited in the calcium

portion of the bone and some other elements are deposited in the soft tissue of the bone, where blood cells could be formed. The calcium portion is not as radiosensitive as the soft tissue.)

Table 17-4-3 lists values of effective absorbed energy for certain nuclides in various organs. Notice the large values for the bone-seeking actinide chains compared with ^{90}Sr and ^{45}Ca.

TABLE 17-4-3
EFFECTIVE ABSORBED ENERGY FOR VARIOUS NUCLIDES AND CHAINS

Nuclide or Parent of Chain	Organ of Reference	Effective Absorbed Energy [ΣE(RBE)n]
^{3}H	Total body	0.01 MeV
^{14}C	Total body	0.054 MeV
^{32}P	Total body	0.69 MeV
^{45}Ca	Bone	0.43 Mev
	Total body	0.086 MeV
^{60}Co	Total body	1.5 MeV
	Lung	0.72 Mev
	Liver	0.72 MeV
	Kidney	0.56 MeV
^{90}Sr	Total body	1.1 MeV
	Bone	5.5 MeV
^{131}I	Thyroid	0.23 MeV
^{226}Ra + dtrs	Bone	110.0 MeV
^{232}Th + dtrs	Bone	270.0 MeV
	Lung	46.0 MeV
^{238}U + dtrs	Bone	220.0 MeV
^{237}Pu + dtrs	Lung	53.0 MeV
	Bone	270.0 MeV

When radioactivity is in the organ, dose produced will be directly proportional to amount. Also to be considered, however, are the distribution of the radioactivity within the organ and the size of the organ. Size of the organ is an important factor, for a given quantity of radioactivity in a small organ will produce a larger dose than the same amount in a larger organ. The reason, of course, is that the same amount of energy is deposited in a larger mass in the case of the larger organ. The size of the organ also enters into the calculation of $\Sigma E(RBE)n$. Ordinarily, it is assumed that radioactivity is uniformly or evenly spread or distributed throughout the organ. This radioactivity initially may be more or less uniformly spread throughout the lungs; but as time passes, it will be cleared from the lungs and translocated, or moved, to the pulmonary lymph nodes in the manner discussed above. A similar case occurs with ingested radioactivity, which may be distributed in various portions of the gastrointestinal tract.

If dose calculations are to be made, size of the organ must be known, along with other characteristics that include biological half-life, breathing rate, and other physiological factors. These factors differ from one person to the next, just as do more obvious physcial differences such as height and weight. Since it is both impractical and difficult (if not impossible) to obtain specific measurements of the masses of various organs and other physiological data from each individual, the scientific community has agreed upon certain values as representative of "standard man." Standard-man values are intended to represent an average adult man, who weighs 70 kg (154 lb) and is 170 cm (5'7") tall. These are worldwide values developed by the International Commission on Radiological Protection, but in the U.S. the

standard man is slightly taller and heavier. In recent years, data for a standard woman, child, and infant also have been developed. Table 17-4-4 lists some important data for the standard man as obtained from ICRP Committee II Report (1959).

TABLE 17-4-4
SYNOPSIS OF STANDARD MAN

	Mass	Effective Radius
Total body	70 kg	30 cm
Muscle	30 kg	30 cm
Skin	6.1 kg	0.1 cm (thickness)
Fat	10 kg	20 cm
Bone	7 kg	5 cm
G-I tract	2 kg	30 cm
Lungs	1 kg	10 cm
Liver	1.7 kg	10 cm
Kidneys	0.3 kg	7 cm
Thyroid	0.02 kg	3 cm
Breathing rate: 2×10^7 cm^3/day (20 liter/min during normal work activity 8 hr per day)		

The data in Table 17-4-4 can be modified slightly to meet special conditions, such as a smaller or larger person. However, all the MPC's in use today are based upon the standard-man values developed by ICRP.

Once necessary information regarding the radionuclide is known (activity, effective absorbed energy, effective half-life, and the mass of the organ), dose delivered to the organ can be calculated simply. Ordinarily, the assumption is made that radioactivity is uniformly concentrated in the organ. Dose rate can be calculated from Equation 17-4-3 for a given and constant quantity of activity.

$$\text{rem/hr} = \frac{kq\varepsilon}{m} \qquad \text{Eq. 17-4-3}$$

where:

k = a constant.
q = activity.
ε = effective absorbed energy.
m = mass of organ.

The constant k is important and will be developed in the following example.

EXAMPLE

Given: Suppose a person had 1 μCi of ^{131}I in the thyroid.
Find: The dose rate to the thyroid.
Solution: From Table 17-4-4, the mass of the thyroid is 20 g, and from Table 17-4-3, ε, effective absorbed

energy is 0.23 MeV. Use Equation 17-4-3:

$$\text{rem/hr} = \frac{kq\varepsilon}{m}$$

$$\text{rem/hr} = \frac{k \times 10^{-6}\ \text{Ci} \times 0.23\ \text{MeV/dis}}{20\ \text{g}}$$

The constant k is required in Equation 17-4-3 to convert units of Ci-MeV per gram-dis to rem/hr and is derived as shown in Equation 17-4-4.

$$k = \frac{2.2 \times 10^{12}\ \text{d/min-Ci} \times 60\ \text{min/hr} \times 1.6 \times 10^{-6}\ \text{erg/Mev}}{100\ \text{erg}}$$

$$k = 2.1 \times 10^{6}\ \frac{\text{dis-g-rem}}{\text{Ci-MeV-hr}} \qquad \text{Eq. 17-4-4}$$

When combined, Equations 17-4-3 and 17-4-4 produce:

$$\text{rem/hr} = \frac{2.1 \times 10^{6} \times 0.23 \times 10^{-6}}{20} = 0.24\ \text{rem/hr}$$

The general equation can be obtained by combining equations 17-4-4 and 17-4-3 to give Equation 17-4-5.

$$\text{rem/hr} = \frac{2.1 \times 10^{6}\ q\varepsilon}{m} \qquad \text{Eq. 17-4-5}$$

Equation 17-4-5 gives only the initial dose rate from the material; if no more is added, that dose rate will be reduced as the radioactivity decays or is removed from the organ. To obtain the dose rate at any time t after the initial time, a decay correction is made. Dose rate to the organ at time t becomes –

$$\text{rem/hr at time } t = \frac{2.1 \times 10\ qE}{m} e^{-\lambda_{eff} t} \qquad \text{Eq. 17-4-6}$$

where:

λ_{eff} = effective decay constant (= $\frac{0.693}{T_{eff}}$).

t = time.

q = activity at the starting, or original, time.

Since dose rate constantly is decreasing, dose over a period of time cannot be calculated without use of calculus. In the case of short-lived materials, however, lifetime dose can be calculated by the division of initial dose rate by effective decay constant, λ_{eff}, which results in Equation 17-4-7.

$$\text{Lifetime dose equivalent (rem)} = \frac{2.1 \times 10^6\ q\varepsilon}{m\lambda_{eff}} \qquad \text{Eq. 17-4-7}$$

Note that λ_{eff} must be in units of hours^{-1}.

Equation 17-4-7 now can be used to calcualte lifetime dose equivalent from 1 μCi of ^{131}I in the thyroid. As in the Example, ε = 0.23 MeV and m = 20 g. The value for λ_{eff} can be calculated easily by first obtaining the effective half-life of ^{131}I. From Table 17-4-2, the biological half-life of iodine is 138 days. Radiological half-life is 8.08 days. The effective half-life can be calculated from Equation 17-4-2.

$$T_{eff} = \frac{T_R \times T_B}{T_R + T_B}$$

$$= \frac{8.08 \times 138}{8.08 + 138}$$

$$T_{eff} = 7.6 \text{ days} = 183 \text{ hours}$$

The effective decay constant is simply $\frac{0.693}{T_{eff}}$.

So, for ^{131}I, λ_{eff} is $\frac{0.693}{183} = 3.8 \times 10^{-3}$ hr.

Equation 17-4-7 now can be solved. (Most MPC calculations involve only exponential loss from the body.)

$$\text{Lifetime dose equivalent} = \frac{2.1 \times 10^{6} \times 10^{-6} \times 0.23}{20 \times 3.8 \times 10^{-3}} = 6.4 \text{ rem.}$$

The equations discussed above serve as the basis for all internal dose calculations. These calculations easily can become mathematically complicated, particularly if power function excretion equations are used, as is the case with a few of the Maximum Permissible Concentrations. MPCs assume a continuous intake over a 40-hour work week in such a manner that the amount of radioactivity in the organ is constant, giving a constant dose equivalent rate (rem/hr) to the organ. This level is, of course, the maximum permissible exposure for the organ. Thus, a dose equivalent rate equal to the maximum permissible exposure is given to the critical organ by exposure to one MPC, 40 hours each week, year after year.

The concept of critical organ is important to radiation protection. Simply stated, the critical organ is the organ that receives the largest fraction of its MPE from a given

quantity of radioactivity in the body. For ^{131}I, the thyroid is the critical organ because most of the ^{131}I in the body will be in that tiny organ. For some elements, there may be more than one critical organ, depending upon route of entry and solubility. ^{90}Sr, for example, ordinarily is taken into the body in soluble form, in which case, bone is the critical organ. If inhaled in insoluble form, the lung becomes the critical organ since enough of the ^{90}Sr remains in the lung to give this organ the greatest fraction of its MPE. If ingested, the gastrointestinal tract is the critical organ because absorption of insoluble forms of ^{90}Sr into the body is very small, and the G-I tract will receive the greatest fraction of MPE in any organ.

Another important concept in internal dose assessment is that of Maximum Permissible Body Burden (MPBB). The MPBB is that quantity of a specific radionuclide that, continuously present <u>in the body</u>, will produce an exposure of 1 MPE to the critical organ. Notice that the words "<u>in the body</u>" have been underlined to emphasize the fact that this amount is <u>not in the critical organ, but in the body as a whole</u>. From the literature, the MPBB for the familiar example ^{131}I is 0.7 μCi. This quantity of activity continuously present in the body will produce a dose of 1 MPE to the thyroid. Dose to other parts of the body will be much less. The MPE for the thyroid has been set at 30 rem/yr or 0.6 rem/week. (Note: The thyroid is a relatively radio-resistant organ and, therefore, has a higher MPE.) With the aid of Equation 17-4-5, the amount of iodine <u>in the thyroid</u> that would produce this dose equivalent rate can be calculated. First, the hourly dose equivalent is computed:

$$\frac{0.6 \text{ rem/wk}}{168 \text{ hr/wk}} = 3.6 \times 10^{-3} \text{ rem/hr} = 3.6 \text{ mrem/hr.}$$

Then Equation 17-4-5 is employed.

$$\text{rem/hr} = \frac{2.1 \times 10^6 \, q\varepsilon}{m}$$

$$3.6 \times 10^{-3} = \frac{2.1 \times 10^6 \times 0.23 \times q}{20}$$

Next, q is found.

$$q = \frac{3.6 \times 10^{-3} \times 20}{2.1 \times 10^6 \times 0.23}$$

$$= 1.5 \times 10^{-7} \text{ Ci}$$

$$q = 0.15 \text{ } \mu\text{Ci}$$

Thus, only 0.15 μCi of ^{131}I in the thyroid produces a dose equivalent rate equal to 1 MPC. But the MPBB for ^{131}I is 0.7 μCi. This suggests that only –

$$\frac{0.15}{0.7} = 0.2 \text{ or } 20\%$$

of the ^{131}I in the body under constant intake conditions is required to maintain a constant level in the thyroid. The remainder is in the blood, kidneys, urine, or other organs.

Assessment of internal dose equivalents to various organs of the body is a difficult task that requires a good knowledge of the physiological characteristics and of the physical data of the specific nuclide. Problems, however, are greatly simplified through use of the standard man and standardized calculation techniques. Although these methods will not be precise, errors are well within acceptable limits for radiation protection work. Indeed, most errors fall on the side of safety and tend, in most cases, to overestimate the dose equivalent delivered. Thus, a small measure of safety is added to the already conservative radiation protection standards.

MPCs and MPBBs are based upon two criteria:

1. For bone-seeking radionuclides, body burden (and hence MPC) is based upon direct comparison with ^{226}Ra, for which human data is available. This criterion is applicable only to nuclides that emit α and β (+ or -) particles. Weekly dose to the bone is 0.6 rem/wk.
2. For all other radionuclides, body burden is based upon the following limits:
   - 0.1 rem/wk to total body and gonads.
   - 0.6 rem/wk to skin and thyroid.
   - 0.3 rem/wk to all other tissues.

MPC calculations in all cases except one are based upon intake of radioactive material by the body. The one exception is when the body is immersed in a cloud or atmosphere that contains a radioactive gas not well absorbed, but merely breathed in and out of the lungs. This is the case with the various inert gases such as Ar and Xe. For these gases only, MPC_a is based upon the dose to skin and external dose. In these cases, the skin is the critical organ.

EXERCISES

1. If the biological half-life for a radionuclide is very long (say, a million years), what effect would there be on dose rate?
2. MPE for bone is 0.6 rem/wk and for lung is 0.3 rem/wk. Determine which is the critical organ for a body containing 50 μCi if 20% is in the lung and 60% is in the bone.

MATERIALS

None.

PROCEDURES

None.

DATA TABLES

None.

REFERENCE MATERIALS

International Commission on Radiological Protection. "Report of Committee II on Permissible Dose from Internal Radiation," Health Physics, 3:1 (1960).

_____. Report 30, "Biological Effects of Inhaled Radionuclides," Annuals of the ICRP, Vol. 2, No. 3 (1979).

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Moe, H.J.; Lasuk, S.R.; Schumacher, M.C.; and Hunt, H.M. Radiation Safety Technician Training Course. ANL-7391, Rev. 1. Prepared for ARGONNE National Laboratory. Springfield, VA: National Technical Information Service, U.S. Department of Commerce, 1972.

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Norwood, W.D. Health Protection of Radiation Workers, Class C. Kaukauna, WI: Thomas Publishing Co., 1975.

1. Define the following terms:
   a. Maximum Permissible Concentration.

   b. Maximum Permissible Body-Burden.

   c. Bone seeker.

   d. Organ dose.

   e. Critical organ.

   f. Biological half-life.

   g. Standard man.

   h. Phagocytosis.

2. Describe the ways in which radioactive material can enter the body.

3. A 62-kg youth inhales 0.1 μCi of an insoluble alpha emitter. Given the following, calculate dose to the critical organ (lungs). Any table in the text may be used.

$\Sigma E(RBE)n$ - 64 MeV.

4. If 90% of the alpha emitter in Test item 3 were removed from the lungs and if the remaining 10% went to pulmonary lymph nodes with a mass of 15 g, what would be the dose to this organ?

5. Calculate permissible lung burden for ^{239}Pu. If 25% of the ^{239}Pu is in the lung, what would MPBB be if the lung is the critical organ?

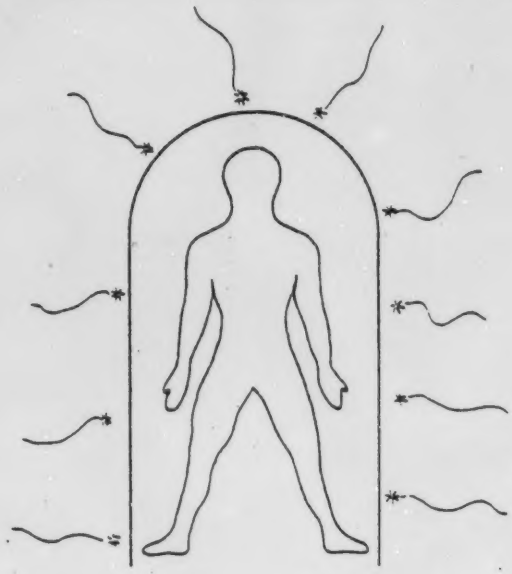
6. The isotope ^{136}I has a half-life of 83 seconds, decaying to ^{136}Xe, which is stable. What might be the critical organ? Why?

7. A quantity of ^{129}I ($T_R = 1.6 \times 10^7$ y) is taken into the thyroid and produces an initial dose rate to that organ of 6 mrem/hr. What is the dose rate one month later? ten years later?

NUCLEAR TECHNOLOGY

MODULE 17 - 5

BIOASSAY TECHNIQUES AND RECORDKEEPING
FOR RADIATION PROTECTION



RADIATION PROTECTION II

COURSE 17

ORD CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

COURSE MODULES

COURSE 17 RADIATION PROTECTION II

MODULE 17-1	Film Dosimetry
MODULE 17-2	Thermoluminescent Dosimetry (TLD)
MODULE 17-3	Radiation Dose Assessment - Personnel Dosimeters
MODULE 17-4	Internal Dose Assessment
MODULE 17-5	Bioassay Techniques and Recordkeeping for Radiation Protection

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INTRODUCTION

In the previous module, techniques of internal dose assesment were presented with one major exception: measurement of the amount of radioactivity in the body. Measurement of the amount of radioactivity in the body is known as "bioassay" and can be accomplished in several ways. This module treats the various methods of bioassay and compares their advantages and disadvantages. It also provides some experiments to demonstrate how bioassay can be performed, and to give the student an idea of the amount of radioactivity normally in the body.

This module also covers record keeping, an apparently unrelated topic, but one of importance to the whole subject of personnel dosimetry and radiation protection in general. Without records of what has transpired before, no radiation protection program could hope to be effective and efficient.

MODULE PREREQUISITES

The student should have completed courses in basic chemistry and physics; Course 1, "Radiation Physics"; Course 2, "Radiation Protection I"; Course 11, "Radiation Detection and Measurement"; and the first four modules in this course.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Identify the various types of bioassay for internal radioactivity, including the strengths and weaknesses of each.
2. Given a sample of urine, perform a gross beta analysis, and calculate the concentration of radioactivity.
3. Measure the concentration of tritium in the body, and determine the amount of activity.
4. Define the following terms:
   - In-vivo counting.
   - Whole-body counting.
   - Radiourinalysis.
   - Reportable exposure.
   - Organ counting.
   - Shadow shield whole-body counter.
   - Phantom.
   - Breath analysis.
5. Explain what is meant by ALARA.
6. State four reasons for the maintenance of personnel dosimetry records.
7. Design a form for the following:
   - Personal exposure report to an individual.
   - Wipe survey.
   - Instrument survey.

BIOASSAY

Bioassay is the measurement of the amount and kind of radioactivity within a person. Bioassay can be divided conveniently into two broad categories:

1. In-vivo counting.
2. Analysis of body fluids and excreta.

Each category has certain advantages and limitations, and, unfortunately, none is useful for all, or even most, radionuclides encountered in medicine, industry, or research.

The most direct and simple bioassay method involves direct counting in which the body itself is the source. Such direct counting is known as in-vivo counting. The words in vivo are Latin and mean "in life," or "alive." In-vivo counting can be either of the whole body or of a portion of the body. A whole-body counter is simply a device with which the amount of radioactivity in the body can be measured.

The basic idea behind in-vivo counting is simple. Radioactive materials inside the body emit radiations – gamma and X-ray – that escape from the body and that can be measured with ordinary radiation monitoring equipment. In theory, this measurement is quite simple and easy to do, but in practice it actually is quite difficult. First, it should be noted again that in-vivo counting works only with radionuclides whose radiations can escape the body. Thus, radioactive substances that emit only alpha or beta radiations – for example, ^{14}C or ^{32}P – usually will not be measurable by in-vivo counting techniques. This is a significant and important limitation to the use of in-vivo counting. Nuclides

such as ^{60}Co, for example, which emit high-energy gamma rays, can be measured by the proper equipment with relative ease.

In-vivo counting involves measurement of the gamma-ray energy coming out of the body. This energy is compared with known energies of gamma rays from various radionuclides. The process is gamma spectroscopy, which can be used not only to identify what the radioactive materials in the body are, but also to determine how much is in the body. Quantitative measurement – determination of how much – is not done easily, for many sources or error can creep into the measurement, as will be discussed below.

In-vivo measurement of the total amount of radioactivity in the body is known as "whole-body counting." Since the amount of radioactivity in the body usually is quite small – perhaps only a few tenths of a microcurie – whole-body counters must be quite sensitive. Usually, the person is placed into a specially shielded room to keep out naturally-occurring background radiations. One or two large detectors, such as NaI(TL) crystals, are utilized to measure gamma rays coming out of the body. The person normally lies on a table or sits in a specially designed chair that provides reproducible geometry. The detector, usually a large scintillator, is pointed at the trunk. A second detector may be placed behind the person to improve sensitivity. A typical whole-body counting setup is illustrated in Figure 17-5-1. Pulses from the detector are sorted electronically according to size in a pulse-height analyzer, which enables the individual who performs the measurement to identify the energy of the radiation, since the energy is proportional to the pulse size. Both the energy and the amount of activity can be determined by the number of pulses of a given size.

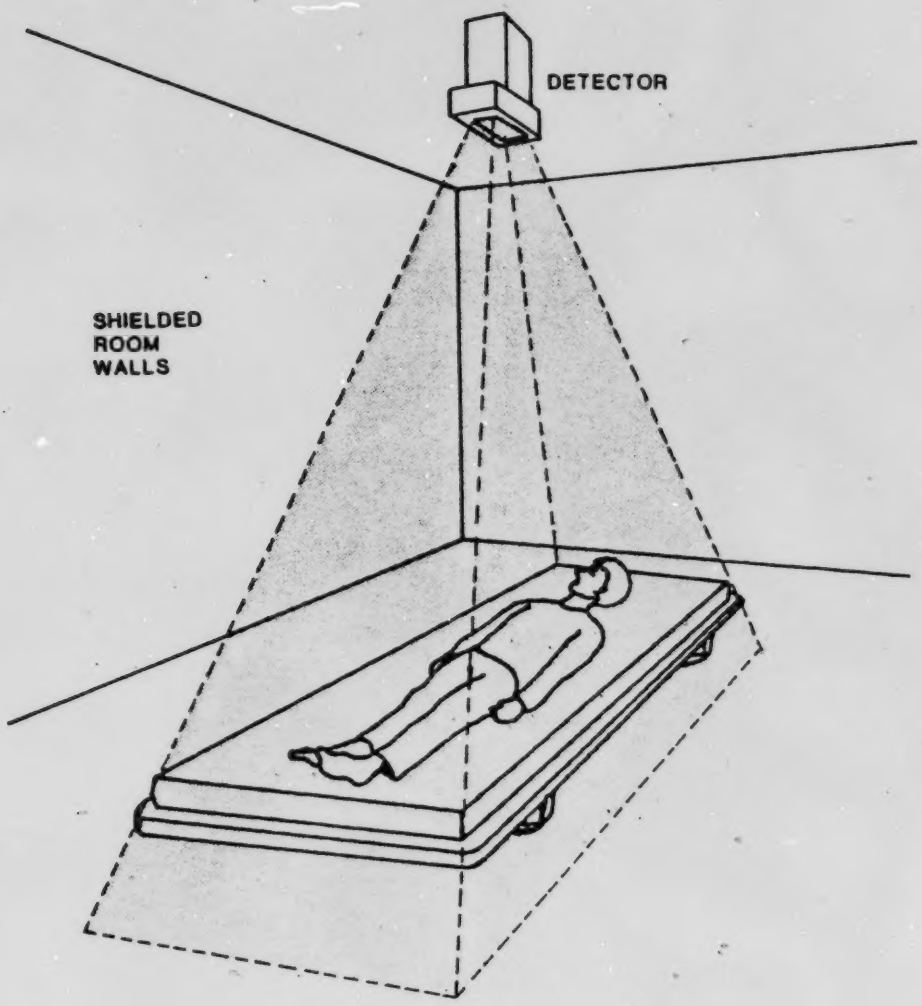

Figure 17-5-1. Schematic representation of a whole-body counter. In practice the detector is placed much closer to the body and hence may not detect activity in the feet.

Although the use of shielded rooms is best, these are expensive, heavy, and stationary. They cannot be moved easily, if at all. To get around these problems, a smaller

version, known as the "shadow shield whole-body counter," is available (Figure 17-5-2). Shadow shield counters are not as sensitive as counters operated in a full shielded room, but they are adequate for most of the gamma-emitting nuclides found in industry, and they are far cheaper to build. Because they are smaller, they can be placed into a truck and transported from place to place.

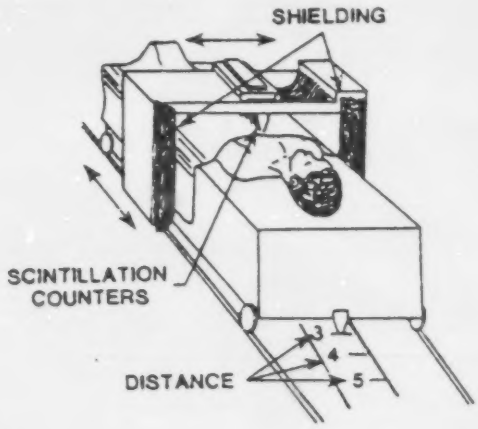

Figure 17-5-2. "Shadow shield" whole-body counter. The subject is moved through the shielding under the scintillation detectors.

Whole-body counting suffers from certain errors and disadvantages. The level of sensitivity must be low enough to detect radionuclides in quantities equal to a small fraction of the MPBB. This level requires not only shielding to reduce background, but also relatively long counting times. Practically, however, a ten- to twenty-minute counting time is employed because longer counts reduce the total number of people who can be examined in a day. Longer counts also are a problem in that the person being counted must remain still during the count. Even at 10 minutes for a count, only four to five people, at most, can be examined in an hour. Time must be allowed between each count to get people settled and to print the accumulated data.

Another drawback to whole-body counting related to reduction of sensitivity by absorption or attenuation of gamma rays within the body. This is not too significant for ^{60}Co, which emits two fairly high-energy gammas (1.17 and 1.33 MeV) with each decay, but it is with ^{241}Am, which emits a 60-keV gamma ray 36% of the time. The lower-energy gamma ray from ^{241}Am is absorbed with relative ease by the tissue through which it must travel to reach the outside of the body; for example, a thickness of only 3.4 cm will reduce the 60-keV gammas from ^{241}Am by half. For ^{60}Co, a thickness of 11.6 cm is required. This effect, known as "self-shielding," can be quite great if the radioactivity is deep within the person, and if the person is quite stout. Moreover, if the material is in the bone, the self-shielding effect will be even greater; the half-value thickness for ^{241}Am gamma rays in bone is less than 1 cm.

Whole-body counting works reasonably well for gamma-emitting radionuclides distributed throughout the body; however, radioactive elements will localize in various tissues of organs as pointed out in Module 17-4, "Internal Dose Assessment." Thus, a scan may be done to determine where the radioactivity is located. Scanning is accomplished by moving the person slowly under the detector to ascertain where the radioactivity is located in the body. The results of a scan are shown in Figure 17-5-3, and a scanner is shown in Figure 17-5-4.

When the location of the radioactivity within the body has been determined by the scan, the radioelements and the organ in which they may be localized can be identified. If the radioactivity seems concentrated in the neck, the reasonable assumption may be made that radioiodine has been deposited in the thyroid gland. A different type of counter then

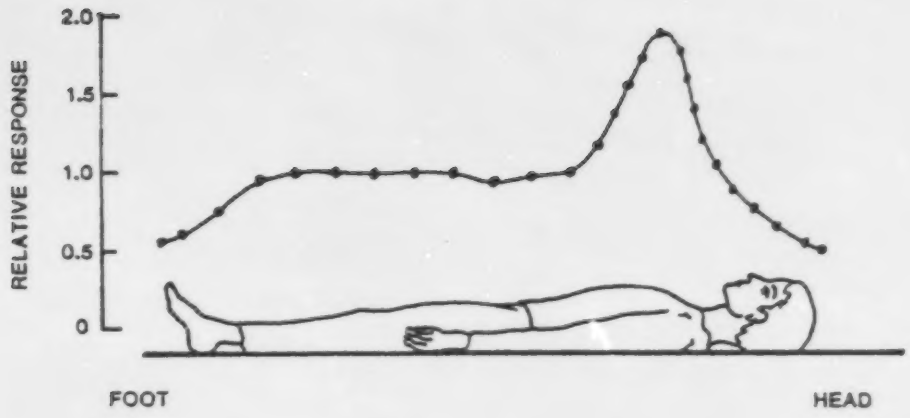

Figure 17-5-3. Results of a scan that indicate activity localized in the chest region.

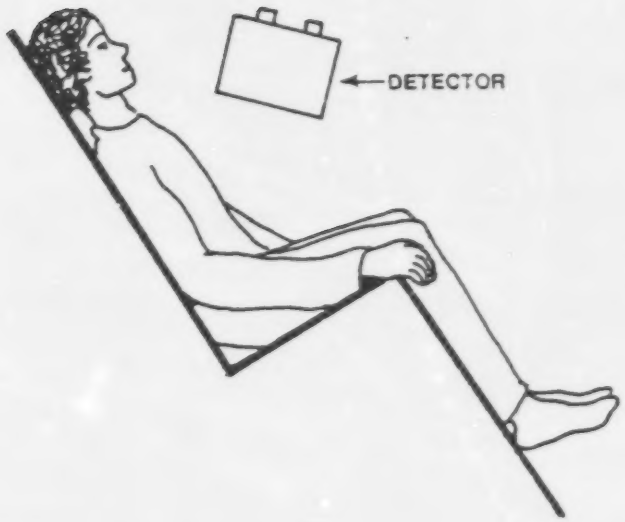

Figure 17-5-4. The so-called "I meter arc" whole-body counting geometry.

can provide a more accurate measurement of the radioactivity. In the case of the example (radioiodine in the thyroid gland), a special scintillation detector may be used. This detector

is well shielded on all sides by a lead pipe or cylindrical shield into which it is placed. The one open (unshielded) end is pointed at the thyroid gland in the neck, and a count is made. A multichannel analyzer is utilized in conjunction with the scintillation detector to identify the specific radionuclide and to estimate how much radioactivity may be in the thyroid gland.

Determination of how much radioactivity is in any particular organ or any part of the body is very difficult to accomplish accurately. Characteristics of the detector, such as efficiency, must be known. Corrections also must be made for geometry, for absorption in the organ itself, and for absorption of radiation in the tissues lying between the organ and the outside of the body. These corrections can be very complicated and difficult in the case of the lungs and certain other organs. A portion of the lungs is protected, or covered, by bone. The bone in this case is the rib cage, which from a shielding or absorption standpoint might be thought of as alternate bands of bone and muscle. Calculation of the shielding is, thus, very difficult.

Sometimes a "phantom," or artificial body, can be used for calibration purposes. A simple phantom can be made by the placing of a human skeleton inside a thin plastic shell formed in the shape of a person. The plastic shell then can be filled with water that contains a known amount of radioactivity to provide a calibration source. More complex phantoms contain plastic shells formed in the shape of individual organs, which can be filled with solutions that contain known amounts of radioactivity to provide a realistic calibration source.

The technique of measuring radioactivity in an organ by in-vivo counting has been given the special name of "organ

counting." Thus, "in-vivo" counting is a special term that can mean either whole-body counting (WBC) or organ counting. The terms often are used incorrectly, however, and care should be taken in the reading of reports and books to ensure that the meaning is understood clearly.

One type of in-vivo counting sometimes is employed in accident situations. This type is wound counting, in which radioactivity deposited in a cut or puncture of the skin is measured with a counter. Wound counting can establish how effective the decontamination of a contaminated wound has been and can guide the health physics and medical staff in determining if surgery or other medical procedures need to be taken to remove the radioactivity.

Because in-vivo counting essentially is useful only for gamma-emitting radionuclides and, in addition, requires considerable equipment and specialized, complex techniques, other kinds of bioassay frequently are used. These involve radiochemical analysis of body wastes and fluids. Most commonly, analyses are made of urine and feces since most radioactivity excreted from the body will be eliminated via the kidneys or gastrointestinal tract. Occasionally, breath is collected and analyzed; less frequently, sweat, saliva, hair, and nails.

Breath analysis is highly specific and usually is employed to determine the amount of ^{226}Ra in bone. Once deposited in bone, the radium remains in the body more or less forever, undergoing alpha decay to ^{222}Rn, a radioactive gas. Some of this gas, which has a 3.8-day half-life, escapes from the bone into the blood and is carried to the lungs, where it diffuses out into the breath. The amount of ^{222}Rn (radon) in the breath is proportional to the amount of radium in the body, and is measured from breath in a balloon

inflated by the person being tested. This special balloon is held for awhile to allow the radon to decay to its daughters, as indicated by the chain decay scheme below, and then is counted for either beta or alpha activity.

$$^{222}Rn \xrightarrow[3.8\ days]{\alpha} {}^{218}Po \xrightarrow[3.1\ min]{\alpha} {}^{214}Bi \xrightarrow[26.8\ min]{\beta}$$

$$^{214}Po \xrightarrow[2 \times 10^{-4}\ sec]{\alpha} {}^{210}Pb \quad (21\ y)$$

Analysis of urine for radioactivity is known as "radiourinalysis." A sample of urine is collected and analyzed by standard chemical and radiological techniques to determine how much radioactivity it contains and to obtain the concentration in μCi/ml, or similar units. There is always radioactivity in urine because of natural radioactivity and fallout from weapons tests. Thus, normal urine contains some tritium (^{3}H), ^{40}K, and ^{137}Cs. A baseline, or background, radiourinalysis can be made on a radiation worker before that individual begins to work with radioactivity. A baseline analysis sometimes is done for gross (total) beta and/or gross alpha radioactivity and sometimes by gamma spectroscopy.

Radiourinalysis is very useful for soluble radioactive materials that have been taken into the body. These materials normally are excreted through the urine via the kidneys. In the event of an exposure or suspected exposure, urine can be collected and analyzed each day to ascertain excretion rate and, hence, the biological and effective half-lives. These values allow dose to be calculated.

Collection of urine on a daily basis may be impractical; therefore, a 24-hour or "spot" (i.e., one time) sample may be collected. Spot sampling is done for routine measurements of the radioactivity in radiation workers. If radioactive contamination of the body is found, additional samples may be collected and analyzed.

Insoluble materials, whether inhaled or ingested, are excreted through the feces. Fecal analysis is much more difficult than urinalysis, and it very seldom is performed on a routine basis. Usually, fecal analysis is performed when an intake of radioactive materials in insoluble form has taken place or is suspected. The feces are collected and then ashed to dryness. The ashing procedure may require several days, but usually can be done in one day. Ashing must be done slowly and with special care in order to prevent loss of the volatile radioactivity.

After an individual has experienced an intake of radioactive material, say by inhalation, both fecal and urine analyses may be required to determine how much was in the body and to calculate the dose. This is particularly important for highly radioactive alpha emitters such as plutonium-239. Figure 17-5-5 displays excretion curves for a partly soluble radioactive material inhaled in a one-time exposure. A portion is cleared very rapidly from the body in the feces; another portion is cleared much more slowly, requiring about a week to be cleared totally.

Activity in the urine also was initially very high and then leveled off after about two weeks. A power function equation can be obtained by combining these curves.

Special preparation techniques frequently are required for fecal and urine samples. These samples contain protein, sodium, chlorine, and other substances that chemically may

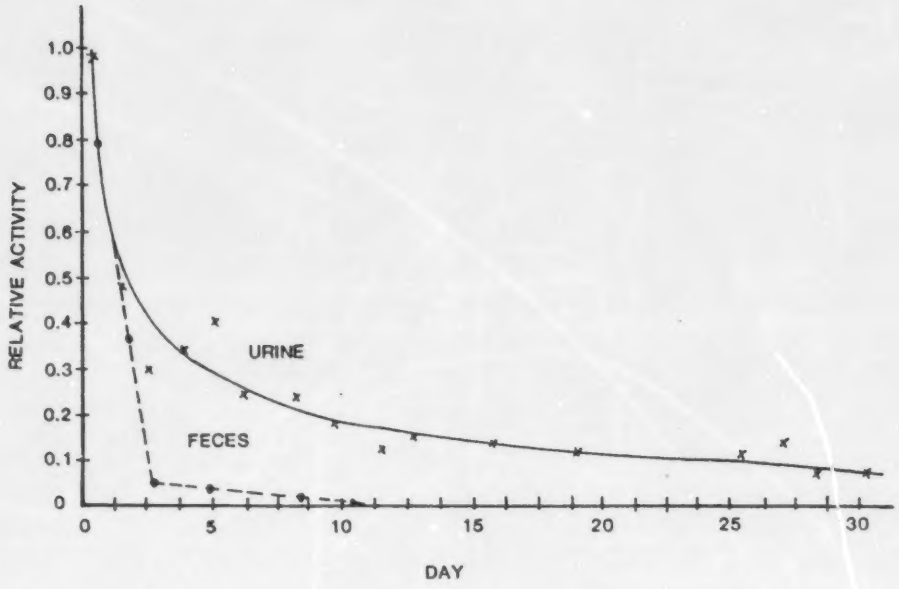

Figure 17-5-5. Clearance curves for a radioactive material inhaled on Day 0.

tie up, or otherwise affect, removal of various radioactive elements by ordinary chemical means. Removal of the radioactive materials – that is, certain specific ones – may be important, since in this fashion they are concentrated and the sensitivity of the method is increased greatly.

RADIATION RECORDS

The gathering and collection of records of dosimeter and bioassay results, radiation surveys, calibration data for instruments, and, in fact, all information related to radiation protection, is an important task. Many records must be kept by law or regulation, especially those records that relate to radiation doses. Others are kept as a protection against unwarranted lawsuits, or to be analyzed and used

to determine ways to keep radiation exposures low. They also are kept for medical reasons and to provide a scientific data base for various studies of radiation workers.

A very important use for radiation records is in an ALARA program. ALARA stands for As Low As Reasonably Achievable, and simply means that radiation exposures to people are to be kept at the lowest practicable level. Exposure limits are just that – limits of the maximum amounts to which people can be exposed. Exposure records are vital to an ALARA program. Doses received by various groups of workers can be compared from one time period to the next, and the records can be used to discover which groups are receiving the largest exposure and where control efforts should be provided. Notice that one of the key words that makes up the ALARA phrase is low and that another is reasonable. The key is to get closer to zero exposure and to do so by practical means. To stop the work or to spend millions of dollars to reduce a dose received by one person from 500 millirem to 450 millirem may not be practicable or reasonable. But addition of some shielding or the change of a source's location may be. Thus, to a large extent, ALARA is based upon judgment, and exact numbers or measures may not apply.

Some of the reasons for proper record keeping can be summarized as follows:

- ALARA, already discussed.
- Legally required.
- Legal protection.
- Medical records.
- Research.

Legal requirements for radiation exposure recordkeeping are given in Title 10, Code of Federal Regulations; in certain

state regulations; and, in some cases, are included in license conditions. Specific requirements, contained in Parts 19 and 20 of Title 10 are discussed below.

Only two sections of Part 19 deal with records, and one of these is indirect. Paragraph 13 (the full citation would be given as 10 CFR 19.13) is quoted on the following pages in its entirety, and requires that individuals be given reports of their radiation exposures by their employers. This paragraph also refers to records that must be kept by the employer. The following paragraph, 10 CFR 19.14, also mentions records, noting that these must be open to inspection by the Nuclear Regulatory Commission.

> Sec. 19.13 Notifications and reports to individuals. — (a) Radiation exposure data for an individual, and the results of any measurements, analyses, and calculations of radioactive material deposited or retained in the body of an individual, shall be reported to the individual as specified in this section. The information reported shall include data and results obtained pursuant to Commission regulations, orders or license conditions, as shown in records maintained by the licensee pursuant to Commission regulations. Each notification and report shall: be in writing; include appropriate identifying data such as the name of the licensee, the name of the individual, the individual's social security number; include the individual's exposure information; and contain the following statement:
>
> > This report is furnished to you under the provisions of the Nuclear Regulatory Commission regulation 10 CFR Part 19. You should preserve this report for further reference.
>
> (b) At the request of any worker, each licensee shall advise worker annually of the worker's exposure to radiation or radioactive materials as shown in records maintained by the licensee pursuant to §20.401 (a) and (c).

(c) At the request of a worker formerly engaged in licensed activities controlled by the licensee, each licensee shall furnish to the worker a report of the worker's exposure to radiation or radioactive material. Such report shall be furnished within 30 days from the time the request is made, or within 30 days after the exposure of the individual has been determined by the licensee, whichever is later; shall cover, within the period of time specified in the request, each calendar quarter in which the worker's activities involved exposure to radiation from radioactive materials licensed by the Commission; and shall include the dates and locations of licensed activities in which the worker participated during this period.

(d) When a licensee is required pursuant to §20.405 or §20.408 of this chapter to report to the Commission any exposure of an individual to radiation or radioactive material the licensee shall also provide the individual a report on his exposure data included therein. Such report shall be transmitted at a time not later than the transmittal to the Commission.

(e) At the request of a worker who is terminating employment in a given calendar quarter with the licensee in work involving radiation dose, or of a worker who, while employed by another person, is terminating assignment to work involving radiation dose in the licensee's facility in that calendar quarter, each licensee shall provide to each such worker, or to the worker's designee, at termination, a written report regarding the radiation dose received by that worker from operations of the licensee during that specifically identified calendar quarter or fraction thereof, or provide a written estimate of that dose if the finally determined personnel monitoring results are not available at that time. Estimated doses shall be clearly indicated as such.

[Sec. 19.13 as amended effective March 3, 1975 (40 F.R. 8783); amended June 6, 1979, effective August 20, 1979 (44 F.R. 32349).]

Sec. 19.14 Presence of representatives of licensees and workers during inspections. – (a) Each licensee shall afford to the Commission at all reasonable times opportunity to inspect materials, activities, facilities, premises, and records pursuant to the regulations in this chapter.

The major record-keeping requirements are contained in Part 20, which requires that records be kept of exposures to personnel who are required to be monitored. In addition, 10 CFR 20.401 – 409 requires certain reports to be made to the NRC. Information required in these reports necessitates the keeping of certain records. Most states have record-keeping requirements very similar to those in the Code of Federal Regulations. For reference, the section of 10 CFR 20 dealing with records and reports is reproduced on the following pages. These regulatory requirements should be read with an eye towards understanding the intent, rather than the letter, of the regulations. And, it should be clearly understood that what the regulations require is only the minimum that needs to be done.

RECORDS, REPORTS, AND NOTIFICATION

[¶6330]

Sec. 20.401. Records of surveys, radiation monitoring, and disposal. – (a) Each licensee shall maintain records showing the radiation exposures of all individuals for whom personnel monitoring is required under Sec. 20.202 of the regulations in this part. Such records shall be kept on Form NRC-5, in accordance with the instructions contained in that form or on clear and legible records containing all the information required by Form NRC-5. The doses entered on the forms or records shall be for periods of time not exceeding one calendar quarter.

(b) Each licensee shall maintain records in the same units used in this part, showing the results of surveys required by §20.201(b), monitoring required by §§20.205(b) and 20.205(c), and disposals made under §§20.302, 20.303, and 20.304.

(c)(1) Records of individual exposure to radiation and to radioactive material which must be main-

tained pursuant to the provision of paragraph (a) of this section and records of bioassays, including results of whole body counting examinations, made pursuant to §20.108, shall be preserved until the Commission authorizes disposition.

(2) Records of the results of surveys and monitoring which must be maintained pursuant to paragraph (b) of this section shall be preserved for two years after completion of the survey except that the following records shall be maintained until the Commission authorizes their disposition: (i) records of the results of surveys to determine compliance with §20.103 (a); (ii) in the absence of personnel monitoring data, records of the results of surveys to determine external radiation dose; and (iii) records of the results of surveys used to evaluate the release of radioactive effluents to the environment.

(3) Records of disposal of licensed material made pursuant to §§20.302, 20.303, or 20.304 shall be maintained until the Commission authorizes their disposition.

(4) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations.

(5) If there is a conflict between the Commission's regulations in this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission purusant to §20.501, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

Sec. 20.402. Reports of theft or loss of licensed material. - (a) Each licensee shall report by telephone to the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office

listed in Appendix D, immediately after its occurrence becomes known to the licensee, any loss or theft of licensed material in such quantities and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas.

(b) Each licensee who is required to make a report pursuant to paragraph (a) of this section shall, within thirty (30) days after he learns of the loss or theft, make a report in writing to the appropriate NRC Regional Office listed in Appendix D with copies to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, setting forth the following information:

(1) A description of the licensed material involved, including the kind, quantity, chemical, and physical form;

(2) A description of the circumstances under which the loss or theft occurred;

(3) A statement of disposition or probable disposition of the licensed material involved;

(4) Radiation exposures to individuals, circumstances under which the exposures occurred, and the extent of possible hazard to persons in unrestricted areas;

(5) Actions which have been taken, or will be taken, to recover the material; and

(6) Procedures or measures which have been or will be adopted to prevent a recurrence of the loss or theft of licensed material.

(c) Subsequent to filing the written report the licensee shall also report any substantive additional information on the loss or theft which becomes available to the licensee, within 30 days after he learns of such information.

(d) Any report filed with the Commission pursuant to this section shall be so prepared that names of

individuals who may have received exposure to radiation are stated in a separate part of the report.

Sec. 20.403. Notification of incidents.– (a) Immediate notification. Each licensee shall immediately notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving byproduct, source, or special nuclear material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body or any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $200,000.

(b) Twenty-four hour notification. Each licensee shall within 24 hours notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving licensed material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more of radiation; or

(2) The release of radicactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $2,000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

Sec. 20.404. Report to former employees of exposure to radiation. [Sec. 20.404 was deleted effective September 17, 1973 (38 F.R. 22220).]

Sec. 20.405. Reports of overexposures and excessive levels and concentrations. (a) In addition to any notification required by §20.403, each licensee shall make a report in writing within 30 days to the appropriate NRC Regional Office listed in Appendix D with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, of (1) each exposure of an individual to radiation in excess of the applicable limits in §§20.101 or 20.104(a) or the license, (2) each exposure of an individual to radioactive material in excess of the applicable limits in §§20.103(a)(1), 20.103(a)(2), 20.104(b) or the license; (3) levels of radiation or concentrations of radioactive material in a restricted area in excess of any other applicable limit in the license; (4) any incident for which notification is required by §20.403; and (5) levels of radiation or concentrations of radioactive material (whether or not involving excessive exposure of any individual) in an unrestricted area in excess of ten times any applicable limit set forth in this part or in the license. Each report required under this paragraph shall describe the extent of exposure of individuals to radiation or to radioactive material, including estimates of each individual's exposure as required by paragraph (b) of this section; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence.

(b) Any report filed with the Commission pursuant to this section shall include for each individual ex-

posed the name, social security number, and date of birth, and an estimate of the individual's exposure. The report shall be prepared so that this information is stated in a separate part of the report.

Sec. 20.406. Notice to employees of exposure to radiation.—[Sec. 20.406 was deleted August 17, 1973, effective September 17, 1973 (38 FR 22220).]

Sec. 20.407. Personnel monitoring reports.—Each person described in §20.408 of this part shall, within the first quarter of each calendar year, submit to the Director of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, the reports specified in paragraphs (a)and (b) of this section covering the preceding calendar year.[1] All other persons specifically licensed by the Commission shall, within the first quarter of calendar years 1979 and 1980, submit to the Director of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar years 1978 and 1979. [1,2]

(a) A report of either (1) the total number of individuals for whom personnel monitoring was required under §§20.202(a) or 34.33(a) of this chapter during the calendar year; or (2) the total number of individuals for whom personnel monitoring was provided during the calendar year: Provided, however, That such total includes at least the number of individuals required to be reported under paragraph (a)(1) of this section. The report shall indicate whether it is submitted in accordance with paragraph (a)(1) or (a)(2) of this section. If personnel monitoring was not required to be provided to any individual by the licensee under §§20.202(a) or 34.33(a) of this chapter during the calendar year, the licensee shall submit a negative report indicating that such personnel monitoring was not required.

(b) A statistical summary report of the personnel monitoring information recorded by the licensee for individuals for whom personnel monitoring was either required or provided, as described in paragraph (a) of this section, indicating the number of individuals whose total whole body exposure recorded during the

previous calendar year was in each of the following estimated exposure ranges:

---

[1]A licensee whose license expires or terminates prior to, or on the last day of the calendar year, shall submit reports at the expiration or termination of the license, covering that part of the year during which the license was in effect.

[2]The Commission will evaluate the data obtained for 1978 and 1979 pursuant to this paragraph, and the benefits derived therefrom and may take action, including publication of notice of proposed rulemaking, to extend or otherwise modify this reporting requirement.

Estimated Whole Body Exposure Range (Rems)[3]	Number of Individuals in Each Range
No measurable exposure...	
Measurable exposure less than 0.1...	
0.1 to 0.25...	
0.25 to 0.5...	
0.5 to 0.75...	
0.75 to 1...	
1 to 2...	
2 to 3...	
3 to 4...	
4 to 5...	
5 to 6...	
6 to 7...	
7 to 8...	
8 to 9...	
9 to 10...	
10 to 11...	
11 to 12...	
12 + ...	

The low exposure range data are required in order to obtain better information about the exposures actually

recorded. This section does not require improved measurements.

Sec. 20.408. Reports of personnel monitoring on termination of employment or work. – (a) This section applies to each person licensed by the Commission to:

(1) Operate a nuclear reactor designed to produce electrical or heat energy pursuant to Sec. 50.21(b) or Sec. 50.22 of this chapter or a testing facility as defined in Sec. 50.2(r) of this chapter;

(2) Possess or use byproduct material for purposes of radiography pursuant to Parts 30 and 34 of this chapter;

---

[3]Individual values exactly equal to the values separating Exposure Ranges shall be reported in the higher range.

(3) Possess or use at any one time, for purposes of fuel processing, fabrication, or reprocessing, special nuclear material in a quantity exceeding 5,000 grams of contained uranium-235, uranium-233, or plutonium or any combination thereof pursuant to Part 70 of this chapter; or

(4) Possess or use at any one time, for processing or manufacturing for distribution pursuant to Part 30, 32 or 33 of this chapter, byproduct material in quantities exceeding any one of the following quantities:

Radionuclide	Quantity in curies
Cesium-137	1
Cobalt-60	1
Gold-198	100
Iodine-131	1
Iridium-192	10
Krypton-85	1,000
Promethium-147	10
Technetium-99m	1,000

(b) When an individual terminates employment with a licensee described in parapgraph (a) of this section, or an individual assigned to work in such a licensee's facility but not employed by the licensee, completes the work assignment in the licensee's facility, the licensee shall furnish to the Director of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, a report of the individual's exposures to radiation and radioactive material, incurred during the period of employment or work assignment in the licensee's facility, containing information recorded by the licensee pursuant to §§20.401(a) and 20.108. Such a report shall be furnished within 30 days after the exposure of the individual has been determined by the licensee or 90 days after the date of termination of employment or work assignment, whichever is earlier.

Sec. 20.409. Notifications and reports to individuals. – (a) Requirements for notifications and reports to individuals of exposure to radiation or radioactive material are specified in §19.13 of this chapter.

(b) When a licensee is required pursuant to §20.405 or §20.408 to report to the Commission any exposure of an individual to radiation or radioactive material, the licensee shall also notify the individual. Such notice shall be transmitted at a time not later than the transmittal to the Commission, and shall comply with the provisions of §19.13(a) of this chapter.

In practice, considerably more records are kept than required by the regulations. A good records program will include information on all health physics activities, including, but not limited to, dosimetry data, field measurements and air sampling data, calibration records, and accident investigations. Most radiation safety records will be retained for many years – perhaps 50 or more. The sheer bulk and enormous numbers or records may result in micro-

filming and computerization to make the records-keeping process more efficient. Duplicate records may be required in order that the information will not be lost in the event that one set is destroyed by fire, earthquake, flood, or other disaster.

Dosimetry records are perhaps the key single type of information retained for radiation protection purposes. These records, by their very nature, contain a great deal of personal information about individuals and, therefore, must be restricted from general access. Dosimetry records usually are considered to be medical records and are treated much in the same way with which a doctor or hospital handles the records of patients. A good dosimetry records system would include the following information:

- Name.
- Social security number.
- Payroll number.
- Date of birth.
- Place of birth.
- Sex.
- Race.
- Military service.
- Prior employment.
- Previous occupational exposure to radiation or radiomimetic drugs (drugs that produce action and effects similar to radiation).
- Medical therapeutic radiation exposure history.
- Unusual diagnostic radiation exposure history.
- Present employment history, including job title(s), work location(s), and work restriction(s).
- Personnel dosimetry results, both from internal and external sources, including dosimeter readings,

bioassay baseline data, exposure calculations, accidental exposure data, and specific excretion curves.

- Any other data that might have a bearing on the employee's personal exposure situation.

As noted in the regulations, each employee must be provided with a periodic report of his or her own radiation exposures. This report is mandatory on an annual basis if so requested by the worker. In actual practice, most employers provide workers with such information without being asked; larger employers use computers to print out forms which are given to each employee. Typically, a form will contain the following information:

- Employee name.
- Employee number.
- Social security number.
- Quarterly dose. } external
- Annual dose. } external
- Lifetime dose. } external
- Internal dose.
- Permissible dose limits.

Field records are another important category of radiation protection records, and include such items as survey records, air sampling and analysis data, wipe test results, and radiation protection logbooks. Logbooks can be very important and useful legal records. Thus, they should be kept in permanently bound books whose pages are numbered in sequence. Writing should be in ink, with no erasures. Errors can be lined out, but still should be readable. Entries always should be signed and dated.

Records keeping is made easier by use of specially designed forms. Thus, a radiation survey form or a wipe testing form might contain a map of the room or building in which the work was being done. Locations at which measurements were made then can be marked easily, quickly, and accurately. All entries should be done in ink, and the form should have a place for the following information:

- Date.
- Time.
- Instruments used and serial number (or other identifying number).
- Name of person(s) performing the work.
- Signature of person filling out the form.

There are many other records kept for radiation protection purposes, but the basic principles are the same as for dosimetry and field records. Clarity and legibility are extremely important, and mistakes can be very costly both in time and trouble.

EXERCISES

1. Design a form for the following purposes:
   a. Annual report of exposure to an employee.
   b. Wipe test for the laboratory.
   c. Instrument survey of the laboratory.
2. Develop a set of criteria for determining which students in the class should be selected for bioassay.

MATERIALS

Clean 100-ml and 500-ml beakers and flasks.
Hot plate.
Evaporating planchets.
Standard solution of ^{32}P. (The instructor will handle this.)
Liquid scintillation cocktail (dioxane).
Liquid scintillation counting system.
Laboratory counting system (scaler, timer, end window GM in shield, α detector).
Pipettes (5 ml)(10 ml).
Graduate cylinder.
Dropper apparatus for evaporation.
Counting calibration sources for α, β, and liquid scintillation systems.

PROCEDURES

TRITIUM IN URINE

1. Take a clean 500-ml beaker, and collect a sample of urine.
2. Obtain a background count on the liquid scintillation counting system of at least 1000 counts, for 20 minutes. Record values in Data Table 17-5-1.
3. Using the calibration sources, determine efficiency of the counters. Record.
4. Take the urine sample, and remove a 5-ml aliquot with a pipette.

5. Mix the urine sample aliquot with 10 ml of dioxane scintillation cocktail. Count the sample for 100 minutes.
6. Count the liquid scintillation standard for 1 to 5 minutes. Record results in Data Table 17-5-2.
7. Calculate amount of tritium in urine. Use the equations in Data Table 17-5-2.
8. Compare results with those obtained from a sample of tap water.
9. Compare the average for the whole class.

GROSS ALPHA AND BETA ACTIVITY IN URINE

1. Obtain a fresh urine sample in a clean beaker.
2. Measure the volume in a graduated cylinder.
3. Obtain a background count on the end-window GM counting system. At least 1000 counts should be obtained.
4. Count a standard beta calibration source; determine efficiency of the system.
5. Evaporate urine sample to dryness in a counting planchet, using either the hot plate or heat lamps. This may have to be done overnight with an automatic drop apparatus.
6. Count the planchet for at least 1000 counts.
7. Using Data Table 17-5-2, determine concentration of beta radioactivity in urine.
8. Obtain a background count on the alpha counting system.
9. Count a standard alpha calibration source; determine efficiency of the system.
10. Count the planchet for at least 20 minutes.

11. Using Data Table 17-5-2, determine concentration of alpha radioactivity in urine.
12. Compute average beta and alpha radioactivity concentrations for the entire class.

DETERMINATION OF UNKNOWN BETA ACTIVITY IN URINE

1. Obtain a spiked sample of urine from the instructor. (This sample will contain a known amount of ^{32}O.)
2. Evaporate the sample to dryness.
3. Perform a beta count as in the previous experiment.
4. Determine concentration of beta activity in the spiked sample.
5. Calculate percent error. What are probable causes of the error?

DATA TABLE 17-5-1

EFFICIENCY DETERMINATION

1. $\frac{\text{Background counts}}{\text{Counting time}}$ = background counts per minute

2. $\frac{\text{Calibration source cpm - background cpm}}{\text{Calibration source disintegrations/min}}$ =

Efficiency

DATA TABLE 17-5-2

ACTIVITY CONCENTRATION IN SAMPLE

1. $\frac{\text{Total sample counts}}{\text{counting time}}$ = sample counts per minute

2. $\frac{\text{Sample cpm - background cpm}}{\text{Efficiency}}$ = Sample disintegrations per minute

3. $\frac{\text{Sample disintegrations/minute}}{2.2 \times 10^6}$ = µCi in sample

4. $\frac{\text{µCi in sample}}{\text{volume of sample in ml}}$ = $\frac{\text{µCi}}{\text{ml}}$ in sample

REFERENCE MATERIALS

American National Standards Institute (ANSI). Practice for Occupational Radiation Exposure Records Systems. ANSI Standard N13.6-1966 (R 1972). New York: ANSI, 1972.

Cember, H. Introduction to Health Physics. New York: Pergamon Press, 1969.

Code of Federal Regulations, Title 10, Parts 19 and 20. Washington, D.C.: U.S. Government Printing Office, latest issue and Supplements.

Fitzgerald, J. J. Applied Radiation Protection and Control. New York: Gordon and Breach, 1971.

1. Define the following terms:

   In-vivo counting.

   Whole-body counting.

   Radiourinalysis.

   Reportable exposure.

   Organ counting.

   Shadow shield whole-body counter.

   Phantom.

   Breath analysis.

2. Identify the various types of bioassay for internal radioactivity, including the strengths and weaknesses of each.

3. Explain what is meant by ALARA.

4. State four reasons for the maintenance of personnel dosimetry records.

5. Design a form for the following:

   Personal exposure report to an individual.

   Wipe survey.

   Instrument survey.

NUCLEAR TECHNOLOGY

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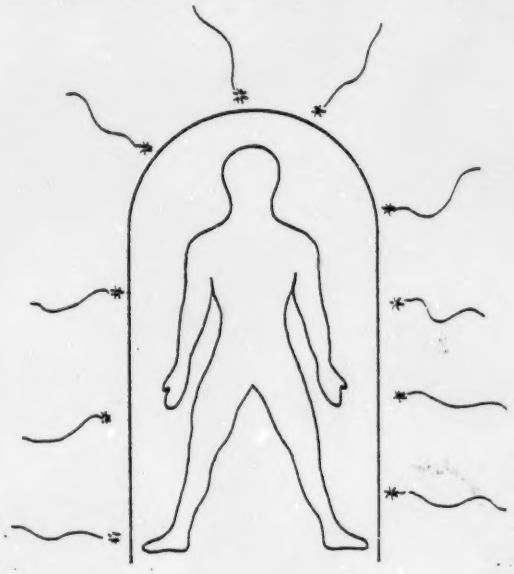
COURSE 17- RADIATION PROTECTION II



RADIATION PROTECTION II

COURSE 17

CENTER FOR OCCUPATIONAL RESEARCH AND DEVELOPMENT

NUCLEAR TECHNOLOGY SERIES

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for

COURSE 17, RADIATION PROTECTION II (FIVE MODULES)

NOTE: Each module in this Course is numbered separately, and the page numbers begin with page 1 in each module. Therefore, this Index lists for each entry the module number (underlined), followed by a comma and the page number. Figures and tables are listed by their numbers, followed by the module and page number in parentheses. A Master Index is available in a separate volume. This general index merges all index entries from the 35 Nuclear Courses. It is a valuable research tool, with which the student or researcher can locate all the references to any subject within the 35 Courses.

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